CHAPTER

44

PWR REACTOR VESSEL ALLOY 600 ISSUES

Jeff Gorman, Steve Hunt, Pete Riccardella, and Glenn A. White

44.1 INTRODUCTION

Primary water stress corrosion cracking (PWSCC) of alloy 600 nickel-chromium-iron base metal and related alloys 82 and 182 weld metal has become an increasing concern for commercial pressurized water reactor (PWR) plants. Cracks and leaks have been discovered in alloys 600/82/182 materials at numerous PWR plant primary coolant system locations, including at several locations in the reactor vessels. The reactor vessel locations include top head control rod drive mechanism (CRDM) nozzles, top head thermocouple nozzles, bottom head instrument nozzles, and reactor vessel outlet and inlet nozzle butt welds. The consequences of this PWSCC have been significant worldwide with 72 leaks through May 2004 (56 CRDM nozzles, 13 reactor vessel closure head thermocouple nozzles, 2 reactor pressure vessel bottom-mounted instrument nozzles, and 1 piping butt weld), many cracked nozzles and welds, expensive inspections, more than 60 heads replaced, several plants with several-month outage extensions to repair leaks, and a plant shutdown for more than 2 years due to extensive corrosion of the vessel head resulting from leak-age from a PWSCC crack in a CRDM nozzle. This chapter addresses alloys 600/82/182 material locations in reactor vessels, operating experience, causes of PWSCC, inspection methods and findings, safety considerations, degradation predictions, repair methods, remedial measures, and strategic planning to address PWSCC at the lowest possible net present value cost.

Several example cases of PWSCC, and resulting boric acid corrosion, are described in the following paragraphs of this chapter and, in some cases, the remedial or repair measures are described. It is important to note that the repairs and remedial measures described may not apply to all situations. Accordingly, it is important to review each new incident on a case-by-case basis to ensure that the appropriate corrective measures are applied, including the need for inspections of other similar locations that may also be affected.

44.2 ALLOY 600 APPLICATIONS

Figure 44.1 shows locations where alloy 600 base metal and alloy 82 or 182 weld metal are used in PWR plant reactor vessels. It should be noted that not all PWR reactor vessels have alloys 600/82/182 materials at each of the locations shown in Fig. 44.1.

44.2.1 Alloy 600 Base Metal

Alloy 600 is a nickel-based alloy (72% Ni minimum, 14–17% Cr, 6–10% Fe) with high general corrosion resistance that has been widely used in light water reactor (LWR) power plants, i.e., in PWRs and boiling water reactors (BWRs). In PWR plants, alloy 600 has been used for steam generator tubes, CRDM nozzles, pressurizer heater sleeves, instrument nozzles, and similar applications. The alloy was originally developed by the International Nickel Corporation (INCO) and is also known as Inconel 600, which is a trademark now held by the Special Metals Corporation [1]. The reasons that alloy 600 was selected for use in LWRs in the 1950s and 1960s include the following [2–7]:

(a) It has good mechanical properties, similar to those of austenitic stainless steels.
(b) It can be formed into tubes, pipes, bars, forgings, and castings suitable for use in power plant equipment.
(c) It is weldable to itself and can also be welded to carbon, low-alloy, and austenitic stainless steels.
(d) It is a single-phase alloy that does not require postweld heat treatment. Also, when subjected to postweld heat treatments that are required for low-alloy steel parts to which it is welded, the resulting sensitization (decreased chromium levels at grain boundaries associated with deposition of chromium carbides at the boundaries) does not result in the high susceptibility to chloride attack exhibited by austenitic stainless steels that are exposed to such heat treatments.
(e) It has good general corrosion resistance in high temperature water environments, resulting in low levels of corrosion products entering the coolant and resulting in low rates of wall thinning.
(f) It is highly resistant to chloride stress corrosion cracking (SCC), and has better resistance to caustic SCC than austenitic stainless steels.
(g) Its thermal expansion properties lie between those of carbon/low-alloy steels and austenitic stainless steels, making it a good transition metal between these materials.

It was alloy 600's high resistance to SCC, especially chloride-induced SCC, that led to its selection for steam generator tubing in PWRs in the 1950s and 1960s. Several early PWRs had experienced SCC of austenitic stainless steel steam generator tubing, variously attributed to chlorides and caustics, and this had led to a desire to use a tubing alloy with increased resistance to these

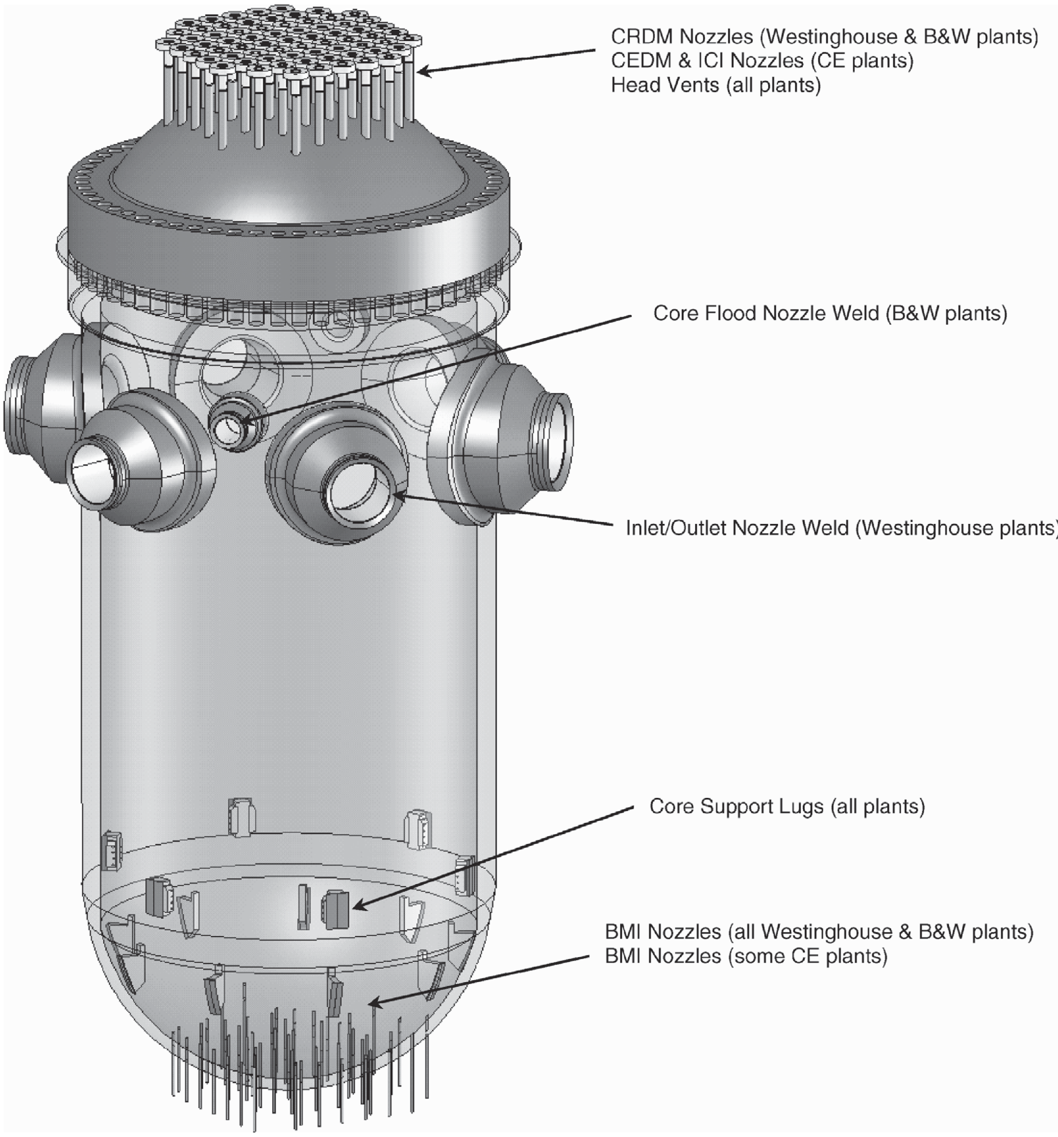

FIG. 44.1 LOCATIONS WITH ALLOYS 600/82/182 MATERIALS IN TYPICAL PWR VESSEL

environments. Similarly, some early cases of SCC of stainless steel nozzle materials in BWRs during initial plant construction and startup, which was attributed to exposure to chlorides and fluorides, led to the wide-scale adoption of alloy 600 and its related weld materials for use in BWR vessel nozzles and similar applications [8].

The first report of SCC of alloy 600 in high-temperature pure or primary water environments was that of Coriou and colleagues in 1959 [9] at a test temperature of 350°C (662°F). This type of cracking came to be known as pure water or primary water SCC (PWSCC) or, more recently, as low potential SCC (LPSCC). In response to Coriou's 1959 report of PWSCC, research was conducted to assess alloy 600's susceptibility to SCC in high-temperature pure and primary water. Most of the results of this research in the 1960s indicated that alloy 600 was not susceptible unless specific contaminants were present [10–12]. The conditions leading to susceptibility included the presence of crevices and the presence of oxygen. Most of the test results of the 1960s did not indicate susceptibility in noncontaminated PWR primary coolant environments. However, by the early 1970s, it had been confirmed by several organizations in addition to Coriou that PWSCC of highly stressed alloy 600 could occur in noncontaminated high-temperature pure and primary water environments after long periods of time [13–15]. Starting with Siemens in the late 1960s, some designers began to move away from use of alloy 600 to other alloys, such as alloy 800 for steam generator tubes and austenitic stainless steels for structural applications [15]. By the mid-1980s, alloy 690, an alternate nickel-based alloy with about twice as much chromium as alloy 600 (~30% vs. ~15%), had been developed and began to be used in lieu of alloy 600 for steam generator tubing [16]. By the early 1990s, alloy 690 began to be used for structural applications such as CRDM nozzles and steam generator divider plates.

44.2.2 Alloys 82 and 182 Weld Metal

Weld alloys 82 and 182 have been commonly used to weld alloy 600 to itself and to other materials. These alloys are also used for nickel-based alloy weld deposit (buttering) on weld preparations and for cladding on areas such as the insides of reactor vessel nozzles and steam generator tubesheets. Alloy 82 is bare electrode material and is used for gas tungsten arc welding (GTAW), also known as tungsten inert gas (TIG) welding. Alloy 182 is a coated electrode material and is used in shielded metal arc welding (SMAW). The compositions of the two alloys are somewhat different, leading to different susceptibilities to PWSCC. Alloy 182 has lower chromium (13–17%) than alloy 82 (18–22%) and has higher susceptibility to PWSCC, apparently as a result of the lower chromium content. Most welds, even if initiated or completed with alloy 82 material, have some alloy 182 material.

In recent years, alloys 52 and 152, which have about 30% chromium and are thus highly resistant to PWSCC, have been used in lieu of alloys 82 and 182, respectively, for repairs and for new parts such as replacement reactor vessel heads.

44.2.3 RPV Top-Head Penetrations

CRDMs in Westinghouse- and Babcock & Wilcox–designed PWR plants and control element drive mechanisms (CEDMs) in Combustion Engineering–designed PWR plants are mounted on the top surface of the removable reactor vessel head. Figure 44.2 shows a typical CRDM nozzle in a Babcock & Wilcox-designed plant. Early vintage Westinghouse PWR plants have as few as 37 CRDM nozzles and later vintage Combustion Engineering plants have as many as 97 CEDM nozzles. These nozzles are machined from alloy 600 base metal with finished outside diameters ranging from 3.5 to 4.3 in. and with wall thicknesses ranging from about 0.4 to 0.8 in. In some cases, a stainless steel flange is welded to the alloy 600 nozzle with an alloy 82/182 butt weld. The nozzles are installed in the reactor vessel head with a small clearance or interference fit (0.004 in. maximum interference on the diameter) and are then welded to the vessel head by an alloy 82/182 J-groove weld. The surface of the J-groove weld preparation is coated with a thin butter layer of alloy 182 weld metal before stress relieving the vessel head so that the nozzles can be installed and the final J-groove weld can be made after vessel stress relief. This avoids possible distortion that could occur if the CRDM nozzles were welded into the vessel head before vessel stress relief.

Most vessels have a single 1.0–1.3 in. outside diameter alloy 600 head vent nozzle welded to a point near the top of the head by a J-groove weld. Two of the early Babcock & Wilcox–designed vessels had eight 1.0-in. outside diameter alloy 600 thermocouple nozzles welded to the periphery of the head by J-groove welds.

Most of the Combustion Engineering vessels have alloy 600 incore instrument (ICI) nozzles welded to the periphery of the top head by J-groove welds. These ICI nozzles are similar to CEDM nozzles except that they range from 4.5 to 6.6 in. outside diameter. Several Westinghouse plants have 3.5 to 5.4 in. outside diameter alloy 600 auxiliary head adapters and de-gas line nozzles attached to the top head by J-groove welds. Several Westinghouse plants have 5.3 to 6.5 in. outside diameter internals support housings and auxiliary head adapters attached to the vessel top head surface by alloy 82/182 butt welds.

In summary, PWR reactor vessels have 38 to 102 alloy 600 nozzles welded to the top head, with most of these attached to the heads after stress relief of the head by alloy 82/182 J-groove welds.

44.2.4 BMI Penetrations

All of the Westinghouse and Babcock & Wilcox–designed reactor vessels in the United States and three of the Combustion Engineering–designed reactor vessels in the United States have alloy 600 instrument nozzles mounted to the vessel bottom heads. These are often referred to as bottom-mounted instrument (BMI) nozzles. These nozzles range from 1.5 to 3.5 in. outside diameter. As shown in Fig. 44.3, a typical BMI nozzle is welded to the bottom head by a J-groove weld. In the case of the Westinghouse and Combustion Engineering plants, the J-groove welds were made after stress relieving the vessel. In the case of the Babcock & Wilcox–designed plants, the J-groove welds were made prior to vessel stress relief. Early test experience at a Babcock & Wilcox-designed plant showed a flow vibration concern with the portions of the BMI nozzles inside the bottom head plenum. Accordingly, all of the Babcock & Wilcock plant BMI nozzles were modified after initial installation to increase the diameter of the portion of the nozzle extending into the lower plenum. The new extension was alloy 600 and the modification weld was made using alloy 82/182 weld metal, with no subsequent stress relief heat treatment.

44.2.5 Butt Welds

Many Westinghouse reactor vessels have alloy 82/182 butt welds between the low-alloy steel reactor vessel inlet and outlet nozzles and the stainless steel reactor coolant pipe, as shown in Fig. 44.4. In most cases, these welds include alloy 182 cladding on the inside of the nozzle and an alloy 182 butter layer applied to the end of the low-alloy steel nozzle prior to vessel stress relief.

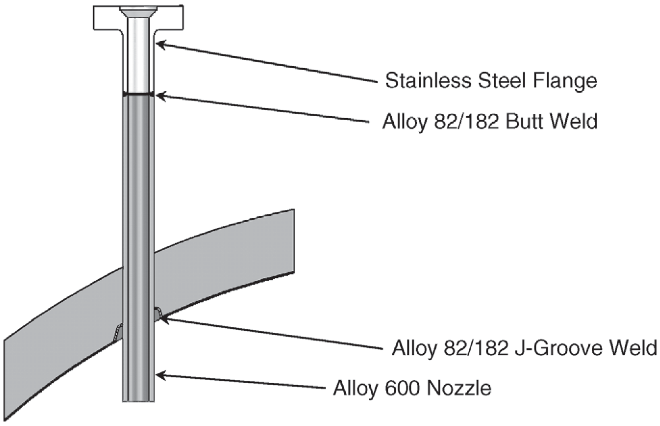

FIG. 44.2 TYPICAL CONTROL ROD DRIVE MECHANISM (CRDM) NOZZLE

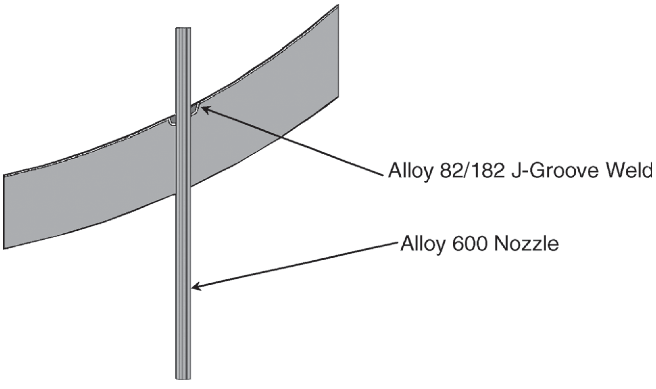

FIG. 44.3 TYPICAL BOTTOM-MOUNTED INSTRUMENT (BMI) NOZZLE

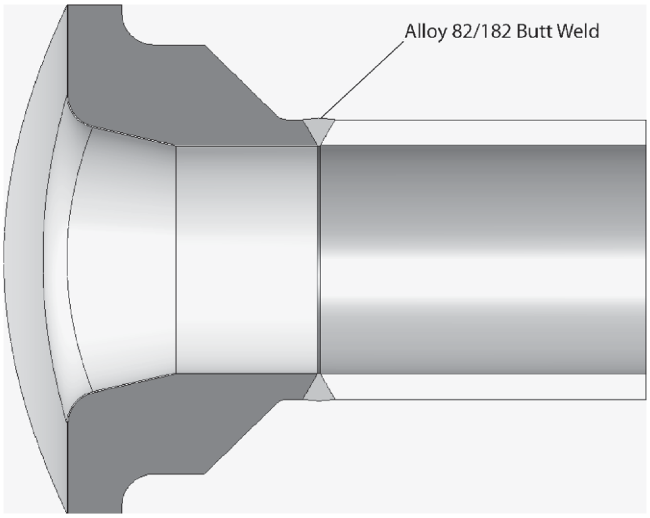

FIG. 44.4 TYPICAL REACTOR VESSEL INLET/OUTLET NOZZLE

Babcock & Wilcox–designed plants, and all but one Combustion–Engineering-designed plant, do not have alloy 82/182 butt welds at reactor vessel inlet and outlet nozzles since the reactor coolant piping is low-alloy steel as opposed to stainless steel.

Reactor vessel core flood line nozzles in Babcock & Wilcox–designed plants have alloy 182 cladding and alloy 82/182 butt welds between the low-alloy steel nozzle and stainless steel core flood pipe.

44.2.6 Core Support Attachments

Most PWR vessels have alloy 600 lugs attached to the inside surface of the vessel, as shown in Fig. 44.5, to guide the reactor internals laterally or to support the reactor internals in the event of structural failure of the internals. These lugs are attached to cladding on the inside of the vessel by full penetration alloy 82/182 welds. In most cases, the vessel cladding in the area of the lugs is also alloy 182 weld metal.

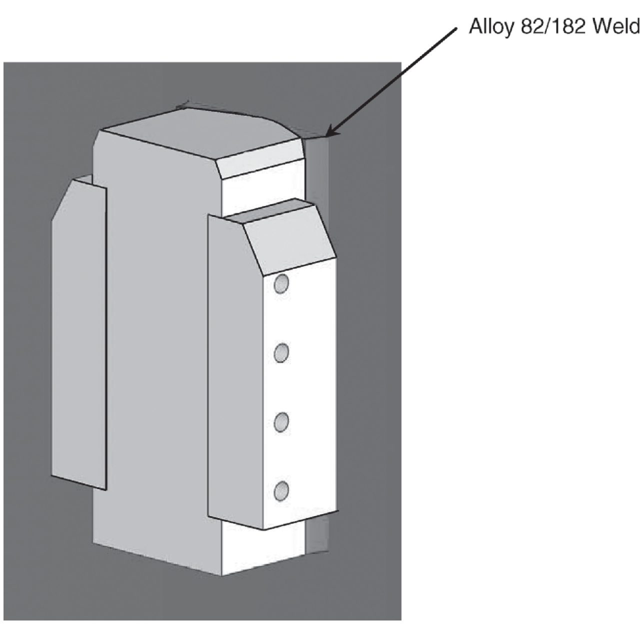

FIG. 44.5 TYPICAL CORE SUPPORT LUG

44.2.7 Miscellaneous Alloy 600 Parts

Most reactor vessel lower closure flanges have alloy 600 leakage monitor tubes welded to the flange surface by alloys 82/182 weld metal. These are not discussed further since the leakage monitor tubes are not normally filled with water and, therefore, are not normally subjected to conditions that contribute to PWSCC.

44.3 PWSCC

44.3.1 Description of PWSCC

PWSCC is the initiation and propagation of intergranular cracks through the material in a seemingly brittle manner, with little or no plastic deformation of the bulk material and without the need for cyclic loading. It generally occurs at stress levels close to the yield strength of the bulk material, but does not involve significant material yielding.

PWSCC occurs when three controlling factors, material susceptibility, tensile stress, and the environment, are sufficiently severe. Increasing the severity of any one or two of the three factors can result in PWSCC occurring, even if the severity of the remaining factor or factors is not especially high. The three factors are discussed separately in the following sections.

While mechanistic theories for PWSCC have been proposed, a firm understanding of the underlying mechanism of PWSCC has not been developed. Accordingly, the influence of material susceptibility, stresses, and environment must be treated on an empirical basis, without much support from theoretical models.

44.3.2 Causes of PWSCC: Material Susceptibility

Based on laboratory test data and plant experience, the following main factors influence the susceptibility of alloy 600 base metal and its weld alloys to PWSCC:

(a) *Microstructure*. Resistance to PWSCC tends to increase as the fraction of the grain boundaries that are decorated by chromium carbides increases. Various models have been proposed to explain this effect such as one where the carbides act as dislocation sources and enhance plastic deformation at crack tips, thereby blunting the cracks and impeding their growth [17]. The absence of carbides in the matrix of grains also correlates with higher resistance to PWSCC, as does larger grain size [18].

(b) *Yield Strength*. Susceptibility to PWSCC appears to increase as the yield strength increases. However, this is considered to be a result of higher yield strength material supporting higher residual stress levels and is, therefore, more of a stress than a material effect. As discussed in para. 44.3.3, tests indicate that the time to PWSCC initiation varies inversely with the fourth to seventh power of the total (applied plus residual) tensile stress [19–21].

(c) *Chromium Concentration*. Tests of wrought materials and weld materials in the nickel–chromium–iron alloy group of materials consistently indicate that susceptibility to PWSCC decreases as the chromium content increases [22,23]. Materials with 30% chromium or more are highly resistant to PWSCC. The improved resistance of alloy 82 vs. alloy 182 weld metal is attributed to the higher chromium concentration of alloy 82 (18–22%) vs. that of alloy 182 (13–17%). Alloy 690 base metal and alloys, 52 and 152 weld metal, with about 30% chromium, have been found to be highly resistant to PWSCC in numerous tests.

(d) *Concentrations of Other Species and Weld Flaws*. No clear trends in PWSCC susceptibility have been observed as a function of the concentration of other species in the alloy such as carbon, boron, sulfur, phosphorous, or niobium. However, to the extent that these species, in combination with the thermomechanical processing to which the part is subjected, affect the carbide microstructure, they can have an indirect influence on susceptibility to PWSCC. Also, hot cracks caused by some of these species (e.g., sulfur and phosphorous) can act as PWSCC initiators and, thus, increase PWSCC susceptibility.

44.3.3 Causes of PWSCC: Tensile Stresses

Industry design requirements, such as ASME BPVC Section III, specify the allowable stresses for reactor vessel components and attachments. The requirements typically apply to operating condition loadings such as internal pressure, differential thermal expansion, dead weight, and seismic conditions. However, the industry design standards do not typically address residual stresses that can be induced in the parts during fabrication. These residual stresses are often much higher than the operating condition stresses and are ignored by the standards since they are secondary (self-relieving) in nature. It is the combination of operating condition stresses and residual stresses that lead to PWSCC.

For the case of penetrations attached to the vessel heads by partial penetration J-groove welds, high residual stresses are caused by two main factors. Firstly, the surfaces of nozzles are typically machined prior to installation in the vessel. This machining cold works a thin layer (up to about 0.005 in. thick) on the surface, thereby significantly increasing the material yield and tensile strength near the surface. Secondly, weld shrinkage, which occurs when welding the nozzle into the high restraint vessel shell, pulls the nozzle wall outward, thereby creating yield strength level residual hoop stresses in the nozzle base metal and higher strength cold-worked surface layers. These high residual hoop stresses contribute to the initiation of axial PWSCC cracks in the cold-worked surface layer and to the subsequent growth of the axial cracks in the lower strength nozzle base material. The lower frequency of cracking in weld metal relative to base metal may result from the fact that welds tend not to be cold worked and then subjected to high strains after the cold work.

Residual stresses in the nozzles and welds can lead to crack initiation from the inside surface of the nozzle opposite from the weld, from the outside surface of the nozzle near the J-groove weld, or from the surface of the J-groove weld.

Most PWSCC cracks have been axially oriented. This is consistent with results of finite element stress analyses, which predict that the hoop stresses exceed the axial stresses at most locations. However, axial stresses can also be high and circumferential cracks have occurred in a few cases.

For the case of butt welds, the weld shrinkage that occurs as progressive passes are applied from the outside surface produces tensile hoop stresses throughout the weld, axial tensile stresses on the outside weld surface (and often also the inside weld surface), and a region of axial compressive stress near midwall thickness. The hoop stresses can contribute to axial PWSCC cracks in the weld and the axial stresses can contribute to circumferential cracks. Finite element analyses show that the hoop stresses on the wetted inside surface of a butt weld are typically higher than the axial stresses at high stress locations, such that cracks are predicted to be primarily axial in orientation. However, if welds are repaired on the inside surface, or subjected to deep repairs from the outside surface, the residual hoop and axial stresses on the wetted inside surface can both approach the yield strength of the weld metal and can cause circumferential as well as axial cracks.

44.3.4 Causes of PWSCC: Environment

Several environmental parameters affect the rate of PWSCC initiation and growth. Temperature has a very strong effect. The effects of water chemistry variations are not very strong, assuming that the range of chemistry variables is limited to those that are practical for PWR primary coolant, i.e., with the coolant containing an alkali to raise pH above neutral and hydrogen to scavenge oxygen.

(a) *Temperature*. PWSCC is strongly temperature dependent. The activation energy for crack initiation is about 44 kcal/mole for thick section nozzle materials [24] and 50 kcal/mole for thinner cold-worked steam generator tubing material [25]. The activation energy for crack growth is about 31 kcal/mole [26]. Using these values, the relative factors for crack initiation and growth at typical pressurizer and cold leg temperatures of 653°F and 555°F relative to an assumed hot leg temperature of 600°F are given in Table 44.1.

(b) *Hydrogen Concentration*. Tests using crack growth rate specimens have shown that crack growth tends to be a maximum when the hydrogen concentration results in the electrochemical potential being at or close to the potential where the Ni/NiO phase transition occurs [27]. Higher or lower values of hydrogen decrease crack growth rates. This effect can be substantial, with peak crack growth rates in some cases being up to four times faster when the hydrogen concentration is at the value causing peak growth rate as compared to conditions with hydrogen values well away from the peak growth rate value, as shown in Fig. 44.6 [27]. Tests at various temperatures show that the hydrogen concentration for the Ni/NiO transition varies systematically with temperature, and that the hydrogen concentration causing the peak growth rate exhibits a similar trend, with the

TABLE 44.1 FACTORS ON CRACK INITIATION AND GROWTH TIME AT TYPICAL PWR TEMPERATURES

Location and Temperature	Crack Growth Q = 31 kcal/mole	Crack Initiation Thick Section Nozzles Q = 44 kcal/mole	Crack Initiation Thin Wall Tubing Q = 50 kcal/mole
Typical Cold Leg - 555°F	0.31	0.19	0.15
Typical Hot Leg - 600°F	1.00	1.00	1.00
Pressurizer - 653°F	3.54	6.00	7.66

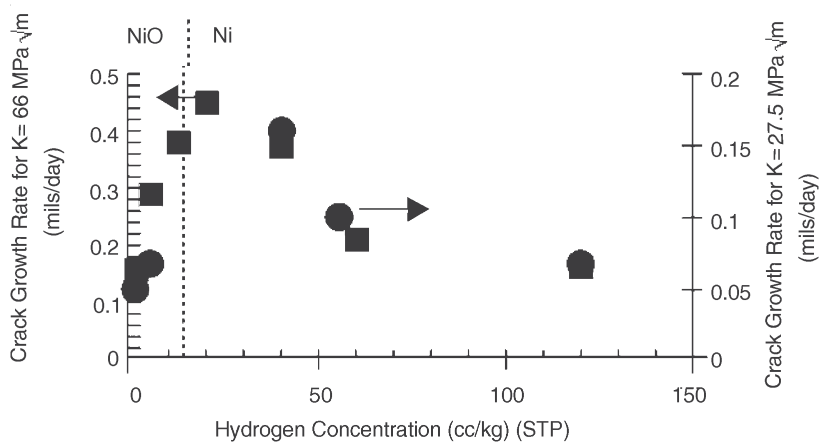

FIG. 44.6 ALLOY 600 CRACK GROWTH RATE AT 338°C PLOTTED VS. HYDROGEN CONCENTRATION [27]

concentration causing the peak crack growth rate becoming lower as temperature decreases (e.g., 10 cc/kg at 320°C, 17 cc/kg at 330ºC, 24 cc/kg at 338°C, and 27.5 cc/kg at 360°C). Crack initiation may depend on hydrogen concentration in a similar manner. However, enough testing to determine the effect of hydrogen on time to crack initiation has only been performed at 330°C, where it resulted in the most rapid crack initiation in alloy 600 tubing at about 32 cc/kg vs. about 17 cc/kg for peak crack growth rate. Reported data regarding effects of hydrogen concentration on PWSCC initiation and growth are shown in Fig. 44.7 [28]. The reasons that the hydrogen concentration for peak aggressivity appears to be about twice as high for crack initiation vs. crack growth rate (32 cc/kg vs. 17 cc/kg) are not known; the difference may be real or may be an artifact of data scatter or imprecision.

(*c*) *Lithium Concentration and pH.* Tests indicate that the effects of changes in pH on crack growth rate, once the pH is well above neutral, are minimal and cannot be distinguished from the effects of data scatter [28]. However, when considering the full pH range from acid to neutral to caustic, several tests indicate that crack growth rates decrease as pH is lowered to the neutral range and below, but is essentially constant for pH_T of about 6 to 8 [29,30].

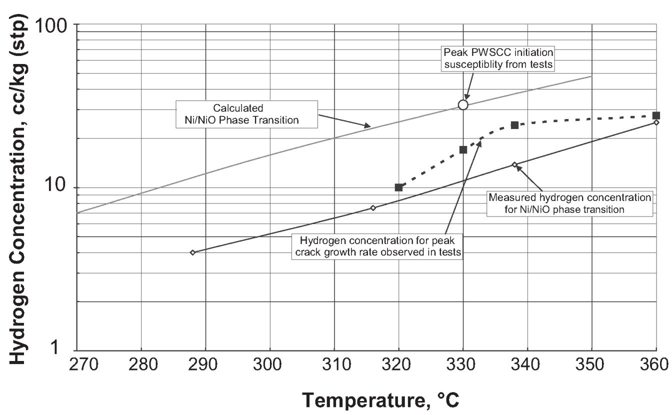

FIG. 44.7 HYDROGEN CONCENTRATION VS. TEMPERATURE FOR N2/N2O PHASE TRANSITION, PEAK PWSCC SUSCEPTIBILITY, AND PEAK CRACK GROWTH RATE [28]

While tests of crack growth rate indicate increases in pH and lithium concentration within the normal ranges used for PWRs have minimal effects on crack growth rate, some evaluations of crack initiation data indicate that increases in pH and lithium cause moderate increases in the rate of crack initiation, e.g., in the range of 10–15% for increases in cycle pH_T from 6.9 to 7.2 [29]. However, recent tests sponsored by the Westinghouse Owners Group (WOG) indicate that the effect may be stronger, such as an increase by a factor of two for an increase in cycle pH_T from 6.9 to 7.2. Further tests under EPRI sponsorship are underway (as of 2004) to clarify this situation.

44.4 OPERATING EXPERIENCE

44.4.1 Precursor PWSCC at Other RCS Locations

PWSCC of alloy 600 material has been an increasing concern in PWR plants since cracks were discovered in the U-bend region of the original Obrigheim steam generators in 1971. The history of PWSCC occurrences around the full reactor coolant system up though 1993, i.e., not limited to the reactor vessel, is documented in an EPRI report [31]. Between 1971 and 1981, PWSCC cracks were detected at additional locations in steam generator tubes (e.g., at dents and at roll transitions), and in an increasing number of tubes. This experience showed that alloy 600 in the metallurgical condition used for steam generator tubes was quite susceptible to PWSCC, with susceptibility increasing as stress, cold work, and temperature increase. It was found that susceptibility was also strongly affected by the microstructure of the material, with susceptibility tending to decrease as the density of carbides on the grain boundaries increases.

The first case of PWSCC of alloy 600 in a non–steam generator tube application was reported in 1982. This incident involved PWSCC of an alloy 600 pressurizer heater sleeve [31]. Swelling of a failed electric heater element inside this sleeve was identified as a contributing cause. Subsequent to this occurrence, an increasing number of alloy 600 instrument nozzles and heater sleeves in pres-surizers have been detected with PWSCC. Also, increasing numbers of instrument nozzles in reactor coolant system hot legs and steam generator heads have also been detected with PWSCC. Many of the susceptible nozzles and sleeves have (as of May 2005) been repaired or replaced on a corrective or preventive basis [31].

PWSCC in alloys 182 and 82 weld metals was first detected in October 2000 in a reactor vessel hot leg nozzle weld [32]. This was only a month before the first detection of PWSCC in a reactor vessel head penetration weld, as discussed in para. 44.4.2.

44.4.2 RPV Top-Head Penetrations

The first reported occurrence of PWSCC in a PWR reactor vessel application involved a leak from a CRDM nozzle at Bugey 3 in France that was detected during a 10-year inservice inspection program hydrostatic test conducted in 1991 [33]. This initial occurrence, and the occurrences detected during the next few years, involved PWSCC of alloy 600 base material at locations with high residual stresses resulting from fabrication. The high residual stresses were mainly the result of weld-induced deformation being imposed on nozzles with cold-worked machined surfaces.

Subsequent to the initial detection of PWSCC in a CRDM nozzle in 1991, increasing numbers of plants detected similar types of PWSCC, typically resulting in small volumes of leakage and boric acid deposits on the head surface as shown in Fig. 44.8. In 2000, circumferential cracks were detected on the outside diameter of some CRDM nozzles. In 2002, significant wastage of the low-alloy steel Davis-Besse reactor vessel head occurred adjacent to an axial PWSCC crack in an alloy 600 CRDM nozzle. The wastage was attributed to corrosion by boric acid in the leaking primary coolant that concentrated on the vessel head. Figure 44.9 shows a photograph of the corroded surface at Davis-Besse. The Davis-Besse plant was shut down for approximately 2 years for installation of a new head and incorporation of changes to preclude similar corrosion in the future. The NRC issued several bulletins describing these events and requiring utilities to document their inspection plans for this type of cracking [34–36].

The cracking discussed above was mainly related to PWSCC of alloy 600 base materials. Starting in November 2000, some plants found PWSCC primarily in the J-groove weld metal, e.g., in CRDM nozzle-to-vessel alloy 182 J-groove welds [37]. Since that time, several other cases of PWSCC of CRDM nozzle-to-head welds have been detected. Also, detection of PWSCC in alloys 182 and 82 welds appears to be increasing in frequency at other non–reactor vessel locations around the reactor coolant system. However, the frequency of PWSCC in welds remains lower than in alloy 600 base material. For example, after the detection of PWSCC in the weld metal of a CRDM nozzle at a PWR in the United States in November 2000, and the detection of PWSCC in the alloy 182 weld metal at reactor vessel outlet nozzles in the United States and Sweden in late 2000, EDF inspected 754 welds in 11 replaced reactor vessel heads without detecting any cracks [24].

44.4.3 RPV Nozzle Butt Welds

In October 2000, a visual inspection showed a leak from an alloys 82/182 butt weld between a low-alloy steel reactor vessel hot-leg outlet nozzle and stainless steel hot-leg pipe at the V.C. Summer plant. Destructive failure analysis showed that the leak was from a through-wall axial crack in the alloys 82/182 butt weld, as shown in Fig. 44.10. The axial crack arrested when it reached the low-alloy steel nozzle on one side and stainless steel pipe on the other side, since PWSCC does not occur in these materials. The axial crack can propagate into the low-alloy steel and stainless steel by fatigue, but the fatigue crack growth rates will be low due to the small number of fatigue cycles. The destructive examination also showed a short-shallow circumferential crack intersecting the through-wall axial crack that grew through alloy 182 cladding and terminated when it reached the low-alloy steel nozzle base metal. Examination of fabrication

FIG. 44.8 TYPICAL SMALL VOLUME OF LEAKAGE FROM CRDM NOZZLE

FIG. 44.9 LARGE VOLUME OF WASTAGE ON DAVIS-BESSE REACTOR VESSEL HEAD

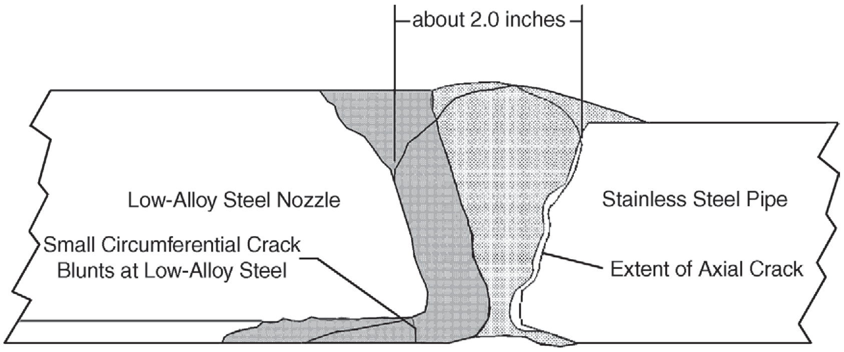

FIG. 44.10 THROUGH-WALL CRACK AND PART-DEPTH CIRCUMFERENTIAL CRACK IN V.C. SUMMER REACTOR VESSEL HOT-LEG OUTLET NOZZLE

records showed that the leaking butt weld had been extensively repaired during fabrication, including repairs made from the inside surface. Nondestructive examinations of other reactor vessel outlet and inlet nozzles at V.C. Summer showed some additional shallow axial cracks.

Shortly before the leak was discovered at V.C. Summer, part-depth axial cracks were discovered in alloys 82/182 reactor vessel outlet nozzle butt welds at Ringhals 3 and 4. Some of these cracks were removed and two were left in place to allow a determination of the crack growth rate. The crack growth rate is discussed in para. 44.7.2.

In addition to the PWSCC cracks in alloys 82 and 182 weld metal in reactor vessel CRDM nozzles and inlet and outlet nozzle butt welds, a leak was found from a pressurizer nozzle butt weld at Tsuruga 2 in Japan and a part-depth crack was detected in a hot-leg pressurizer surge line nozzle butt weld at TMI-1. Both of these cases occurred in 2003. Cracks were also detected in alloys 82 and 182 cladding in steam generator heads that had been hammered and cold worked by a loose part [24].

In the 2005–2008 time period, the industry has begun implementing a massive inspection program for PWSCC in primary coolant loop Alloy 82/182 butt welds (In accordance with Industry Guideline MRP-139 [58] – see Section 44.5.6 below for complete discussion). Considering the temperature sensitivity of the PWSCC phenomenon discussed above, this program started with the highest temperature welds in the system: those at pressurizer nozzles. To date, essentially all pressurizer nozzle dissimilar metal butt welds (typically five or six per plant) have been inspected, mitigated, or both. Approximately 50 nozzles were inspected (many more were mitigated using weld overlays with no pre-inspections), resulting in PWSCC-like indications being detected in nine nozzles, as documented in Table 44.2 below.

Through mid-2008, inspections of reactor vessel nozzle butt welds have not yet been performed; hot leg nozzle inspections under MRP-139 are slated to begin in Fall 2008. Given the above pressurizer nozzle experience, it would not be surprising if at least some welds with PWSCC-like indications are discovered.

TABLE 44.2 CRACKING INDICATIONS DETECTED IN REACTOR COOLANT LOOP ALLOY 82/182 BUTT WELDS, 2005 THROUGH MID-2008

Plant	Inspection Date	Nozzle	Type of Indication	Indication Depth (a, in)	OD Indication Length (l, in)	a / thickness	l / circumference
Calvert Cliffs 2	2005	CL Drain	Circ	0.056	0.628	10%	10%
Calvert Cliffs 2	2005	HL Drain	Axial	0.392	0.000	70%	0%
DC Cook	2005	Safety	Axial	1.232	0.000	88%	0%
Calvert Cliffs 1	2006	HL Drain	Circ	0.100	0.450	19%	5%
Calvert Cliffs 1	2006	Relief	Axial	0.100	0.000	8%	0%
Calvert Cliffs 1	2006	Surge	Circ	0.400	2.400	25%	6%
Davis Besse	2006	CL Drain	Axial	0.056	0.000	7%	0%
San Onofre 2	2006	Safety	Axial	0.420	0.000	30%	0%
San Onofre 2	2006	Safety	Axial	0.420	0.000	30%	0%
Wolf Creek	2006	Relief	Circ	0.340	11.500	25.8%	46%
Wolf Creek	2006	Safety	Circ	0.297	2.500	22.5%	10%
Wolf Creek	2006	Surge	Circ	0.465	8.750	32.1%	19%
Farley 2	2007	Surge	Circ	0.500	3.000	33%	6%
Davis Besse	2008		Axial				
Crystal River 3	2008		Circ				

44.4.4 RPV Bottom-Head Penetrations

In 2003, bare metal visual inspections of the reactor vessel bottom head at South Texas 1 showed small leaks from two BMI nozzles, as shown in Fig. 44.11. These leaks were traced to PWSCC cracks in the nozzles that initiated at small regions of lack-of-fusion in the J-groove welds between the nozzles and vessel head [38]. The nozzles were repaired. Examinations of the other BMI nozzles at South Texas 1 showed no additional cracks. Essentially all other U.S. plants have performed bare metal visual inspections of RPV bottom-head nozzles without any evidence of leaks. At least a dozen U.S. plants have completed volumetric examinations of the BMI nozzles, representing more than 20% of the total population of RPV bottom-head nozzles in the U.S., with no reported cracking. Similarly, no indications of in-service degradation have been identified in volumetric inspections of RPV bottom-head nozzles performed in other countries. PWSCC of BMI nozzles that operate at the plant cold-leg temperature is generally considered to be less likely than PWSCC at locations operating at hot-leg or pressurizer temperatures. The earlier-than-expected PWSCC in BMI nozzles at South Texas 1 may be related to a combination of high material susceptibility and welding flaws.

FIG. 44.11 LEAK FROM SOUTH TEXAS 1 BMI NOZZLE

44.5 INSPECTION METHODS AND REQUIREMENTS

As a result of the increasing frequency of PWSCC cracks and leaks identified in important PWR reactor vessel alloys 600, 82, and 182 materials since 2000, significant efforts are in progress by the nuclear industry and the NRC to improve inspection capabilities and develop appropriate long-term inspection requirements. The following summarizes the status of inspection methods and requirements as of May 2005. It is recommended that users check with the NRC and industry programs to remain abreast of the latest changes in inspection methods and requirements.

44.5.1 Visual Inspections

Bare metal visual inspections have proven to be an effective way of detecting very small leaks, as shown by Figs. 44.8 and 44.11, and, therefore, should play an important role in any inspection program. A key prerequisite for these inspections is that the surface should be free of preexisting boric acid deposits from other sources, because the presence of preexisting boric acid deposits can mask the small volumes of deposits shown in Figs. 44.8 and 44.11. Visual inspections with insulation in place can provide a useful backup to bare metal visual inspections but will be incapable of detecting small volumes of leakage, as shown in Figs. 44.8 and 44.11.

In many cases, it has been necessary to modify insulation packages on the vessel top and bottom heads to facilitate performing bare metal visual inspections. As of May 2005, most of these modifications have been completed for PWR plants in the United States.

ASME Code Case N-722, Additional Examinations for PWR Pressure-Retaining Welds in Class 1 Components Fabricated with Alloys 600/82/182 Materials, Section XI, Division 1, was approved in 2005 to provide for increased visual inspections of potentially susceptible welds for boric acid leakage.

44.5.2 Nondestructive Examinations

Technology exists as of May 2005 to nondestructively examine all of the alloys 600, 82, and 182 locations in the reactor vessel.

Partial penetration nozzles (CRDM, CEDM, ICI) are typically examined using one of two methods. The nozzle base metal can be examined volumetrically from the inside surface by ultrasonics to confirm that the nozzle base material is free of internal axial or circumferential cracks. Alternatively, the wetted surfaces of the alloy 600 base metal and alloys 82 and 182 weld metal can be examined by eddy current probes to ensure that there are no surface cracks. If there are no surface cracks on wetted alloy 600 surfaces, then it can be inferred that there will also be no internal cracks. Nozzles in the reactor vessel top head can be examined when the head is on the storage stand during refueling. Nozzles in the reactor vessel bottom head can be examined ultrasonically or by eddy current when the lower internals are removed from the vessel during a 10-year in-service inspection outage. In some cases, the inside surfaces of BMI instrument nozzles can be examined by tooling inserted through holes in the lower internals.

Reactor vessel inlet and outlet nozzle butt welds are normally inspected ultrasonically from the inside surface using automated equipment. These inspections are typically performed during 10-year in-service inspection outages when the lower internals are removed from the reactor vessel. Eddy current methods are also being used in some cases for examining the inside surfaces of these welds for cracks, although eddy current inspection sensitivity is a function of the condition of the weld surface. For example, discontinuities in the weld profile can cause the eddy current probes to lift off of the surface being examined and, thereby, adversely affect the inspection sensitivity.

CRDM nozzle butt welds can be examined from the outside surface by standard ultrasonic methods.

A key to obtaining good nondestructive examinations is to have the process and the operators qualified on mockups containing prototypical axial and circumferential flaws. The EPRI NDE Center in Charlotte, NC, is coordinating qualification efforts for inspection methods and inspectors in the United States.

44.5.3 ASME BPVC Reactor Vessel Inspection Requirements

ASME BPVC Section XI specifies inservice inspection requirements for operating nuclear power plants in the United States. Portions of these requirements that apply to PWSCC susceptible components in the RPV are summarized as follows:

(a) Table IWB-2500-1, Examination Category B-P, requires a VT-2 visual examination of the reactor vessel pressure-retaining boundary during the system leak test after every refueling outage. No leakage is permitted.
(b) Table IWB-2500-1, Examination Category B-O, requires that 10% of the CRDM nozzle-to-flange welds be inspected by volumetric or surface methods each inspection interval.
(c) Table IWB-2500-1, Examination Category B-N-1, requires that attachment welds to the inside surface of the reactor vessel be examined visually each inspection interval. Welds in the beltline region must be inspected by VT-1 methods while welds outside the beltline region must be inspected by VT-3 methods.
(d) Table IWB-2500-1, Examination Category B-F, specifies examination requirements for dissimilar metal welds in reactor vessels. Nozzle-to–safe end socket welds must be examined by surface methods every inspection interval. Nozzle-to-safe end butt welds less than NPS 4 must be examined by surface methods every inspection interval. Nozzle-to-safe end butt welds NPS 4 and larger must be examined by volumetric and surface examination methods every inspection interval. Some deferrals of these inspections are permitted.
(e) As of May 2005, the ASME Code did not require nondestructive examination of the partial penetration welds for the CRDM and BMI nozzles. However, Code Case N-729-1 [63] was published later in 2005 that contained alternative examination requirements for PWR closure heads with nozzles having pressure-retaining partial-penetration welds. This Code Case included visual, surface and volumetric examinations for PWR closure heads with Alloy 600 nozzles and Alloy 82/182 partial-penetration welds at inspection intervals that are based on the temperature dependence of the PWSCC phenomenon described in para. 44.3.4. (Since RPV closure heads operate at varying temperatures, there are significant head-to-head temperature differences between plants.) Code Case N-729-1 also contains inspection requirements for PWR closure head with nozzles and partial-penetration welds of PWSCC resistant materials to address new and replacement heads.
(f) As noted in para. 44.5.1, Code Case N-722 [64] for visual inspections of alloys 82/182 welds was approved in 2005.
(g) As of May 2008, the ASME Code is working on a new Section XI Code Case that contains alternate inspection requirements Alloys 82/182 welds butt welds. ASME Code actions are also in progress addressing various repair and mitigation options for dealing with PWSCC. These are discussed below in para. 44.9.

44.5.4 NRC Inspection Requirements for RPV Top-Head Nozzles

Subsequent to the discovery of significant corrosion to the Davis-Besse reactor vessel head, the NRC issued NRC Order EA-03-009 [39]. This order specifies inspection requirements for RPV head nozzles based on the effective degradation years of operation. Effective degradation years (EDYs) are the effective full-power years (EFPYs) adjusted to a common 600°F temperature using an activation energy model. For plants with 600°F head temperatures, the EDYs are the same as the EFPYs. For plants with head temperatures, greater than 600°F, the EDYs are greater than the EFPYs. For plants with head temperatures less than 600°F, the EDYs are less than the EFPYs. The NRC order specifies two types of inspections:

(a) bare metal visual inspections of the RPV head surface including 360° around each RPV head penetration nozzle
(b) nondestructive examinations of the RPV nozzles by one of the two following methods:

  (1) ultrasonic testing of each RPV head penetration nozzle (i.e., base metal material) from 2 in. above the J-groove weld to the bottom of the nozzle plus an assessment to determine if leakage has occurred through the interference fit zone
  (2) eddy current testing or dye penetrant testing of the wetted surface of each J-groove weld and RPV head penetration nozzle base material to at least 2 in. above the J-groove weld

The first of the nondestructive examinations is to show that there are no axial or circumferential cracks in the nozzle base metal or leak paths past the J-groove weld. The second of the nondestructive examinations is to show that there are no axial or circumferential cracks in the nozzle base metal by confirming the absence of surface breaking indications on the nozzle and weld wetted surfaces.

The order specifies inspection intervals for three categories of plants: high susceptibility plants with greater than 12 EDY or where PWSCC cracks have already been detected, moderate susceptibility plants less than or equal to 12 EDY and greater than or equal to 8 EDY, and low susceptibility plants with less than 8 EDY.

As of June 2008, the U.S. NRC is expected shortly to transition the requirements for inspection of RPV top-head nozzles based on NRC Order EA-03-009 [39] to a set based on ASME Code Case N-729-1 [63], with caveats. The inspection schedules in this code case are generally based on the RIY (reinspection years) concept, which normalizes operating time between inspections for the effect of head operating temperature using the thermal activation energy appropriate to crack growth in thick-wall alloy 600 material (31 kcal/mol (130 kJ/mol)). The basis for this approach to normalizing for the effect of head temperature is that the time for a flaw just below detectable size to grow to through-wall (and leakage) is dependent on the crack growth process. The requirements in ASME Code Case N-729-1 [63] were developed by ASME, with extensive technical input provided by a U.S. industry group (Materials Reliability Program) managed by EPRI [68].

44.5.5 NRC Inspection Requirements for RPV BMI Nozzles

NRC Bulletin 2003-02, *Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity* [40], summarizes the leakage from BMI nozzles at South Texas 1 and requires utilities to describe the results of BMI nozzle inspections that have been performed at their plants in the past and that will be performed during the next and following refueling outages. If it is not possible to perform bare metal visual examinations, utilities should describe actions that are being made to allow bare metal visual inspections during subsequent outages. If no plans are being made for bare metal visual or nonvisual surface or volumetric examinations, then utilities must provide the bases for concluding that the inspections that have been performed will ensure that applicable regulatory requirements are met and will continue to be met. On September 5, 2003, the NRC issued Temporary Instruction 2515/152 [41], which provides guidance for NRC staff in reviewing utility submittals relative to Bulletin 2003-02. While the Temporary Instruction does not represent NRC requirements, it does indicate the type of information that the NRC is expecting to receive in response to the bulletin.

44.5.6 Industry Inspection Requirements for Dissimilar Metal Butt Welds

The industry in the United States has developed a set of mandatory inspection guidelines for PWSCC susceptible. Alloy 82/182 butt welds, which are documented in the report MRP-139 [58]. MRP-139 defines examination requirements in terms of categories of weldments that are based on 1) the IGSCC resistance of the materials in the original weldment, 2) whether or not mitigation has been performed on the original weldment, 3) whether or not a pre-mitigation UT examination has been performed, 4) the existence (or not) of cracking in the original weldment, and 5) the likelihood of undetected cracking in the original weldment. The categories range from A through K, with the higher letter categories requiring augmented inspection intervals and/or sample size. Category A is the lowest category, consisting of piping that has been replaced (or originally fabricated) with PWSCC resistant material. These weldments are to be inspected at their normal ASME Code frequency, as defined in ASME Section XI, Table IWB-2500-1. Category D refers to unmitigated PWSCC susceptible weld in high temperature locations (e.g. pressurizer or hot leg nozzles). These require an early initial inspection (before end of 2008 for pressurizer nozzles and before 2010 for hot leg nozzles), followed by more frequent inspections if they are not treated with some form of mitigation. Other categories (thru Category K) address susceptible welds that have been mitigated (B and C), welds that have been inspected and found cracked, with or without mitigation, and welds for which geometric or material conditions limit volumetric inspectability. For the latter group, by the time the examination is due, plant owners are required to have a plan in place to address either the susceptibility of the weld or the inspectability of the weld.

At the time of this writing, inspections are well under the MRP-139 guidelines are well underway in U.S. plants. Essentially all pressurizer nozzles have been inspected and or mitigated, and plans are in place to perform the other initial inspections required by MRP-169. Plans include mitigation of most susceptible weldments in high temperature locations, thus moving the weldments into Categories A, B or C. Work is also currently underway to develop an ASME Section XI Code Case (N-790, alternative examination requirements for PWSCC pressure-retaining butt welds in PWRs) which will eventually replace MRP-139 and place the augmented examination requirements for PWSCC susceptible butt welds back under the ASME Section XI Code.

44.6 SAFETY CONSIDERATIONS

44.6.1 Small Leaks

Small leaks due to axial cracks such as shown in Figs. 44.8 and 44.11 do not pose significant safety risk. The leak rates are low enough that the leaking primary coolant water will quickly evaporate leaving behind a residue of dry boric acid. Most of the leaks detected to date have resulted in these relatively benign conditions. As shown in the figures, very small leaks are easily detected by visual inspections of the bare metal surfaces provided that the surfaces are free from boric acid deposits from other sources. One explanation for the extremely low leak rates is that short tight PWSCC cracks can become plugged with crud in the primary coolant, thereby preventing leakage under normal operating conditions. It is hypothesized that distortions, which occur during plant transients, allow small amounts of leakage through the crack before it becomes plugged again. Regardless, these small leaks do not pose a significant safety concern.

44.6.2 Rupture of Critical Size Flaws

Initially, leaking RPV top-head nozzles were thought to be exclusively the result of axial cracks in the nozzles, and it was thus believed that they did not represent a significant safety concern. However, as more examinations were performed, findings arose that called this hypothesis into question.

(a) Relatively long circumferential cracks were observed in two nozzles in the Oconee Unit 2 RPV head, and several other plants also discovered shorter circumferentially oriented cracks.

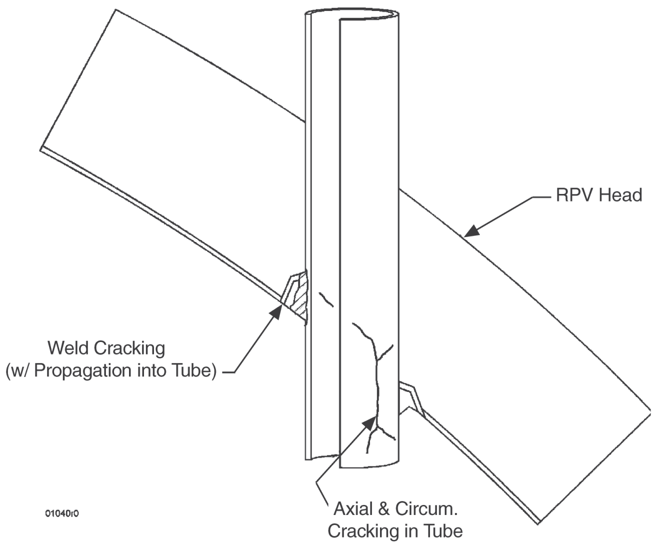

FIG. 44.12 SCHEMATIC OF RPV TOP-HEAD NOZZLE GEOMETRY AND NATURE OF OBSERVED CRACKING

(b) Circumferential NDE indications were discovered in the North Anna Unit 2 head in nozzles that showed no apparent signs of boric acid deposits due to leakage.

Figure 44.12 presents a schematic of a top-head CRDM nozzle and J-groove weld and the nature of the cracking that has been observed. There is some uncertainty as to whether circumferential cracks arise as a result of axial cracks growing through the weld or nozzle and causing leakage into the annular region between the nozzle and head, ultimately leading to reinitiation of circumferential cracking on the outside surface of the tube, or if they are due to the axial cracks branching and reorienting themselves in a circumferential direction, as depicted on the right-hand side of Fig. 44.12. A destructive examination program has been performed on several of the North Anna Unit 2 nozzles, indicating that the circumferential nozzle defects found there were in fact the result of grinding during fabrication rather than service-related cracking. Nevertheless, the occurrence of circumferential cracking adds a new safety perspective to the RPV top-head nozzle cracking problem, because of the potential for such cracks to grow to a critical length and ultimately lead to ejection of a nozzle from the vessel, although a large circumferential flaw covering more than 90% of the wall cross section is typically calculated for nozzle ejection to occur because of the relatively thick wall typical of RPV top-head nozzles.

PWSCC in PWR RPV inlet/outlet nozzles could also potentially develop circumferentially oriented flaws, which could lead to pipe rupture. To date, observed cracking has been primarily axial with only very small circumferential components. With time, however, PWSCC in large piping butt welds might be expected to follow trends similar to the IGSCC cracking issue in BWRs [42]. In the BWR case, cracking and leakage were initially seen only as axially oriented cracks in smaller diameter piping. With time, however, axial and circumferential cracking were observed in pipe sizes up to and including the largest diameter pipes in the system. Considering the potential existence of weld repairs during initial construction of the plants and the associated high residual stresses that they produce in both axial and circumferential directions, significant circumferential cracking may eventually be observed in large-diameter PWR pipe-to-nozzle butt welds.

Because of the concern for PWSCC in PWR piping dissimilar metal butt welds, methods for predicting the critical crack size for rupture in such welds have received recent attention [59]. Axial PWSCC flaws in these welds are limited to the width of the alloy 82/182/132 weld material. Experience has confirmed that the PWSCC cracks arrest when they reach the PWSCC-resistant low-alloy steel and stainless steel materials [50]. Therefore, the maximum axial crack lengths are limited to a few inches at most (much less than the critical axial flaw length), except for the small number of cases involving alloy 600 safe ends or alloy 600 pipe/tube (CRDM and BMI nozzles), where axial cracks initiating in the weld could potentially propagate into the alloy 600 base metal. Thus, critical crack size calculations for PWR piping dissimilar metal butt welds typically assume one or more circumferentially oriented PWSCC flaws.

In 2007, EPRI sponsored a detailed investigation of the growth of circumferential PWSCC flaws in PWR pressurizer nozzle dissimilar metal butt welds [59]. Using finite-element methods, this study examined the effect of an arbitrary crack profile on crack growth and subsequent crack stability and leak rate versus the standard assumption of a semi-elliptical crack profile. The crack stability (i.e., critical crack size) modeling of the EPRI study was based on a standard limit load (i.e., net section collapse) approach as applied to an arbitrary crack profile around the weld circumference [65]. The potential for an EPFM failure mode was considered using a Z-factor approach specific to piping dissimilar metal welds [66]. Finally, the role of secondary piping thermal constraint stresses in the rupture process was investigated on the basis of available experimental pipe rupture data [67], elastic-plastic finite-element analyses of a pipe with an idealized through-thickness crack [59], and pressurizer surge line piping models applied to evaluate the maximum capacity of the secondary loads to produce rotation at a cracked pressurizer surge nozzle [59].

44.6.3 Boric Acid Wastage Due to Larger Leaks

Small concentrations of boron are added to the primary coolant water in PWR plants in the form of boric acid to aid in controlling core reactivity. At the start of an operating cycle with new fuel, the boron concentration is typically about 2,000 ppm or less. The concentration of boron is reduced with fuel burnup to about 0 ppm at the end of an operating cycle when fuel is ready to be replaced. Work by EPRI and others to determine the probable rate of corrosion of low-alloy steel by boric acid is documented in the EPRI Boric Acid Corrosion Guidebook [43]. This document shows that the corrosion rate of low-alloy steel by deareated primary coolant (inside the pressure vessel and piping) with 2,000 ppm boron is negligible. The corrosion rate for low concentration (2,000 ppm) aerated boric acid is also very low. However, when high-temperature borated water leaks onto a hot surface, the water can boil off leaving concentrated aerated boric acid. The corrosion rate of low-alloy steel by hot concentrated aerated boric acid can be as high as 10 in./year under some conditions.

As evidenced by the significant volume of material corroded from the Davis-Besse reactor vessel head, boric acid corrosion has the potential to create significant safety risk. Figure 44.13 shows cross-section and plan views of the corroded region of the Davis-Besse head shown in Fig. 44.9. As indicated, a large volume of the low-alloy head material was corroded, leaving the stainless steel cladding on the inside of the vessel head to resist the internal pressure. Part-depth cracks were discovered in the unsupported section of cladding.

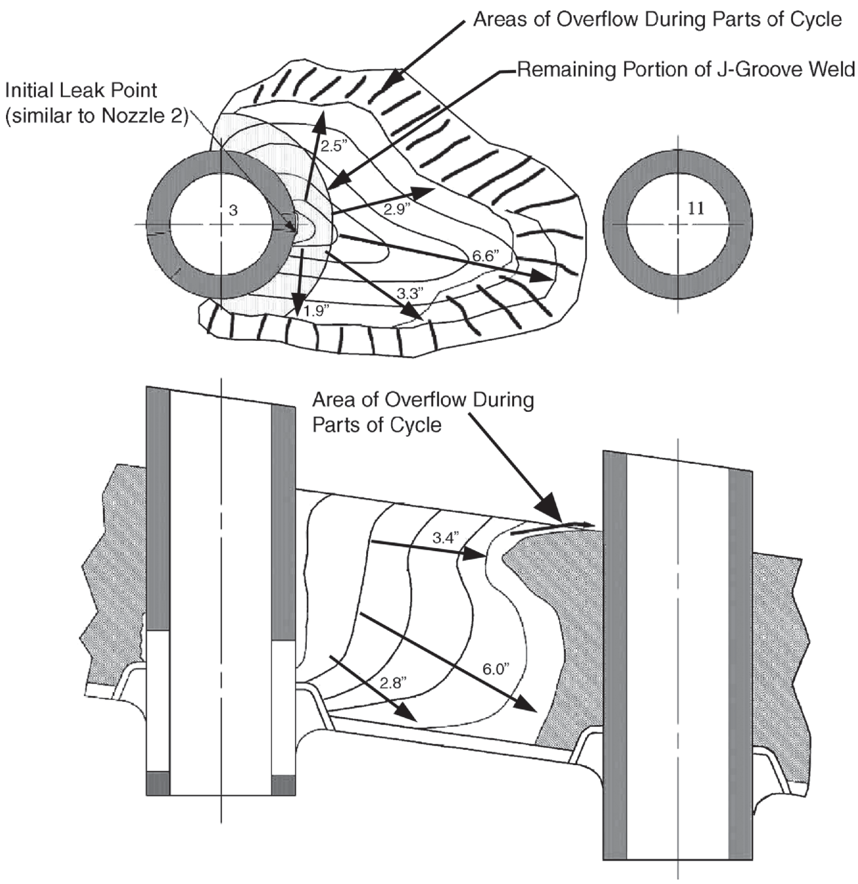

FIG. 44.13 PLAN AND CROSS-SECTION THROUGH CORRODED PART OF DAVIS-BESSE REACTOR VESSEL HEAD

Based on available evidence, it was determined that the leakage that caused the corrosion had been occurring for at least 6 years. While it was known that boric acid deposits were accumulating on the vessel top head surface, the utility attributed the accumulations to leakage from spiral-wound gaskets at the flanged joints between the CRDM nozzles and the CRDMs. The accumulations of boric acid had not been removed due to poor access to the enclosed plenum between the top of the vessel head and the bottom of the insulation, as shown in Fig. 44.14.

The transition from relatively benign conditions, such as shown in Figs. 44.8 and 44.11, to severe conditions, which created the cavity shown in Figs. 44.9 and 44.13, is believed to be a function of the leakage rate. A PWSCC crack that first breaks through the nozzle wall or weld will initially be small (short), resulting in a low leak rate. It is believed that the small leak rate will not lower the metal surface temperature enough to allow liquid conditions to exist. As the crack grows in length above the J-groove weld, the leak rate is expected to increase to the point where boric acid on the surface near the leak remains moist or where the leaking borated water locally cools the low-alloy steel material to the point where the surface will remain wetted and allow boric acid to concentrate. Preliminary models of these conditions have been developed, and test work was started by EPRI in 2004 to more accurately determine the conditions where the leakage produces wetted conditions that can cause high boric acid corrosion rates and where the leakage results in essentially benign dry boric acid deposits.

Conditions such as occurred at Davis-Besse can be prevented by a three-step approach. Firstly, perform nondestructive examinations of the nozzles frequently enough to catch PWSCC cracks before they grow through wall. Secondly, clean the external surfaces of preexisting boric acid deposits from other sources and perform bare metal visual inspections at frequent enough intervals to detect leaks at an early benign stage. Thirdly, if the risk is believed high or

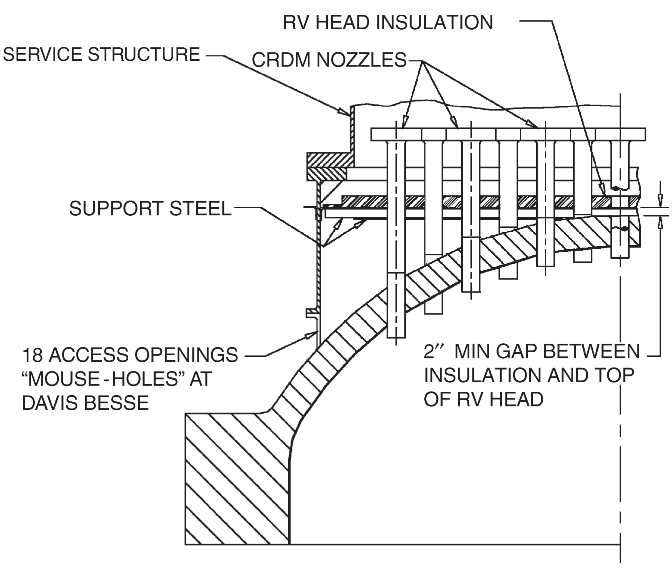

FIG. 44.14 CROSS-SECTION THROUGH DAVIS-BESSE REACTOR VESSEL HEAD

inspections are difficult or costly, replace the susceptible parts or apply a remedial measure to reduce the risk of PWSCC leaks.

44.7 DEGRADATION PREDICTIONS

44.7.1 Crack Initiation

Initiation of PWSCC in laboratory test samples and in PWR steam generator tubing has been found to follow standard statistical distributions such as Weibull and log-normal distributions [44–47]. These distributions have been widely used for modeling and predicting the occurrence of PWSCC in PWRs since about 1988, and continue to be used for this purpose.

The parameters of a statistical distribution used to model a given mode of PWSCC, such as axial cracks in CRDM nozzles, only apply to the homogeneous set of similar items that are exposed to the same environmental and stress conditions, and only to the given crack orientation being modeled. For example, axial and circumferential cracking are modeled separately since the stresses acting on the two crack orientations are different.

In general, two parameter Weibull or log-normal models are used to model and predict the future occurrence of PWSCC. An initiation time, which sometimes is used as a third parameter, is not generally modeled, because use of a third parameter has been found to result in too much flexibility and uncertainty in the predictions.

PWSCC predictions are most reliable when the mode of cracking is well developed with results for detected cracking available for three or more inspections. In this situation, the fitted parameters to the inspection data are used to project into the future. When no cracking has been detected in a plant, rough predictions can still be developed using industry data. This is generally done using a two-step process. The first step involves developing a statistical distribution of times to occurrence of PWSCC at a selected threshold level (such as 0.1%) for a set of plants with similar designs. Data for plants with different temperatures are adjusted to a common temperature using the Arrhenius equation (see Table 44.1). The distribution of times to the threshold level is used to determine a best estimate time for the plant being modeled to develop PWSCC at that threshold level. Techniques are available to adjust the prediction to account for the time already passed at the plant without detecting the mode being evaluated. Once the best estimate time for occurrence at the threshold level is determined, future cracking is projected from that point forward using the median rate of increase (Weibull slope or log-normal standard deviation) in the industry for the mode of PWSCC being evaluated.

44.7.2 Crack Growth

Numerous PWSCC crack growth studies have been performed on thick-wall alloy 600 material in PWR environments at test temperatures that span the range of typical PWR operating temperatures. In 2002, these tests were reviewed and summarized under sponsorship of EPRI [26,48]. The EPRI study (MRP-55) concluded that PWSCC crack growth rates for thick-wall alloy 600 base metal behave in accordance with the following relationship:

$$\dot{a} = \exp\left[-\frac{Q_g}{R}\left(\frac{1}{T} - \frac{1}{T_{ref}}\right)\right]\alpha(K - K_{th})^{\beta}$$

where

$\dot{a}$ = crack growth rate at temperature T in m/sec (or in./hr)
Q_g = thermal activation energy for crack growth
= 130 kJ/mole (31.0 kcal/mole)
R = universal gas constant
= 8.314×10^{-3} kJ/mole • K (1.103×10^{-3} kcal/mole • R)
T = absolute operating temperature at location of crack, K (or R)
T_{ref} = absolute reference temperature used to normalize data
= 325°C = 598.15 K (617°F = 1076.67 R)
α = crack growth amplitude
K = crack tip stress intensity factor, Mpa√m (or ksi√in)
K_{th} = crack tip stress intensity factor threshold
= 9 Mpa√m (8.19 ksi√in)
β = exponent
= 1.16

Temperature dependence is modeled in this crack growth rate equation via an Arrhenius-type relationship using the aforementioned activation energy of 31 kcal/mole. The stress intensity factor dependence is of power law form with exponent 1.16. Figure 44.15 presents the distribution of the coefficient (α) in the power law relationship at constant temperature (617°F). The data in this figure exhibit considerable scatter, with the highest and lowest data points deviating by more than an order of magnitude from the mean. The 75th percentile curve (see Figure 44.15a) was recommended for use in deterministic crack growth analyses [26,48], and this curve is now included in Section XI for disposition of PWSCC flaws in RPV top-head nozzles. In addition, probabilistic crack growth rate studies have been performed of top head nozzles using the complete distribution [49]. An additional factor of 2 has been applied to the 75th percentile value when analyzing crack growth exposed to leakage in the annular gap between the nozzle and the head, to allow for possible abnormal water chemistry conditions that might exist there [26,48].

Similar crack growth rate testing has been conducted for alloys 82 and 182 weld metals. The weld metal crack growth data are sparser and exhibit similar statistical variability. A review of weld metal PWSCC crack growth data has also been completed under EPRI sponsorship [61,62]. This study (MRP-115) showed that Alloy 182/132 weld metal crack growth obeys a similar relationship to that shown above for alloy 600 base metal, but with crack growth rates about four times higher than the alloy 600 curve for stress intensity factors greater than about 20 ksi√in (see Figure 44.15a). Similar to the heat-by-heat analysis for the wrought material, a weld-by-weld analysis was performed on the available worldwide laboratory crack growth rate data for the weld materials (see Figure 44.15b). The EPRI study (MRP-115) concluded that PWSCC crack growth rates for alloy 82/182/132 weld metal behave in accordance with the following relationship, where no credit for a stress intensity factor threshold greater than zero was taken because of insufficient data on this parameter:

$$\dot{a} = \exp\left[-\frac{Q_g}{R}\left(\frac{1}{T} - \frac{1}{T_{ref}}\right)\right]\alpha f_{alloy} f_{orient} K^{\beta}$$

where:

$\dot{a}$ = crack growth rate at temperature T in m/s (or in/h)
Q_g = thermal activation energy for crack growth
= 130 kJ/mole (31.0 kcal/mole)
R = universal gas constant
= 8.314×10^{-3} kJ/mole-K (1.103×10^{-3} kcal/mole-°R)
T = absolute operating temperature at location of crack, K (or °R)

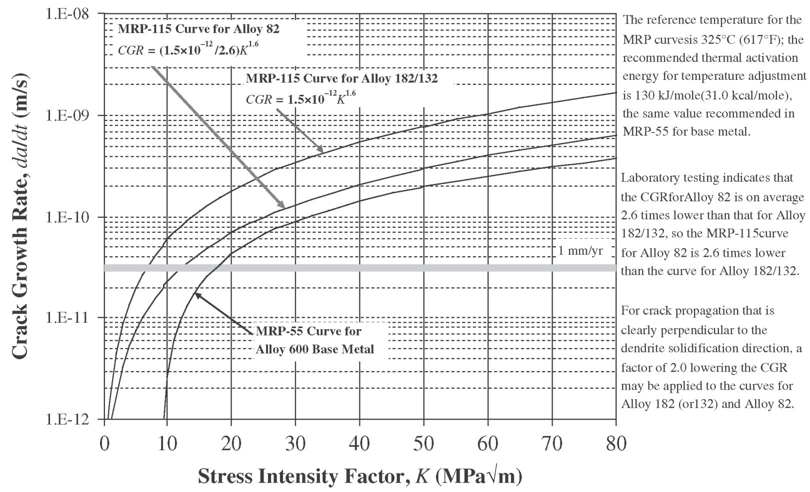

FIGURE 44.15A DETERMINISTIC CRACK GROWTH RATE CURVES FOR THICK-WALL ALLOY 600 WROUGHT MATERIAL AND FOR ALLOY 182/132 AND ALLOY 82 WELD MATERIALS [61,62]

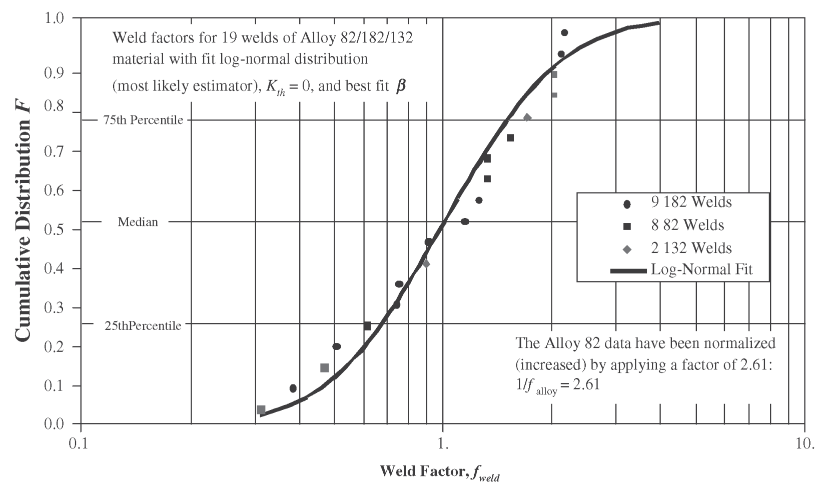

FIGURE 44.15B LOG-NORMAL FIT TO 19 WELD FACTORS FOR SCREENED MRP DATABASE OF CGR DATA FOR ALLOY 82/182/132 [61,62]

T_{ref} = absolute reference temperature used to normalize data
= 598.15 K (1076.67°R)
α = power-law constant
= 1.5×10^{-12} at 325°C for $\dot{a}$ in units of m/s and K in units of MPa √m (2.47×10^{-7} at 617°F for $\dot{a}$ in units of in/h and K in units of ksi √in)
f_{alloy} = 1.0 for Alloy 182 or 132 and 1/2.6 = 0.385 for Alloy 82
f_{orient} = 1.0 except 0.5 for crack propagation that is clearly perpendicular to the dendrite solidification direction
K = crack-tip stress intensity factor, MPa√m (or ksi√in)
β = exponent
= 1.6

Deterministic crack growth rate predictions have been performed for axial and circumferential cracking in RPV top- and bottom-head nozzles and in large-diameter butt welds [49,50]. Welding residual stresses are a primary factor contributing to crack growth in all these analyses. Stress intensity factors versus crack size, considering residual stresses plus operating pressure and thermal stresses are first computed in these studies. These are then inserted into the appropriate crack growth relationship (alloy 600, 82, or 182) at the component operating temperature and integrated with time to predict crack size versus operating time at the applicable temperature.

Figure 44.16 shows typical crack growth predictions for a circumferential crack in a steep angle RPV top-head (CRDM) nozzle. (Nozzles in the outer rings of vessel heads having the steepest angles between the nozzle and the head have been found to be controlling in terms of predicted growth rates for circumferential cracks). The analysis depicted in Fig. 44.16 assumed a through-wall, 30° of circumference crack in the most limiting azimuthal location of the nozzle at time zero, and predicted the operating time for it to grow to a size that would violate ASME Section XI flaw evaluation margins with respect to nozzle ejection (~300°). It is seen that, even for relatively high RPV temperatures, operating times on the order of 8 years or greater are predicted for circumferential nozzle cracks to propagate to a size that would violate ASME Section XI safety margins.

Figure 44.17 shows similar crack growth predictions for a postulated circumferential crack in a large-diameter nozzle butt weld. Stress intensity factors were computed in this analysis for

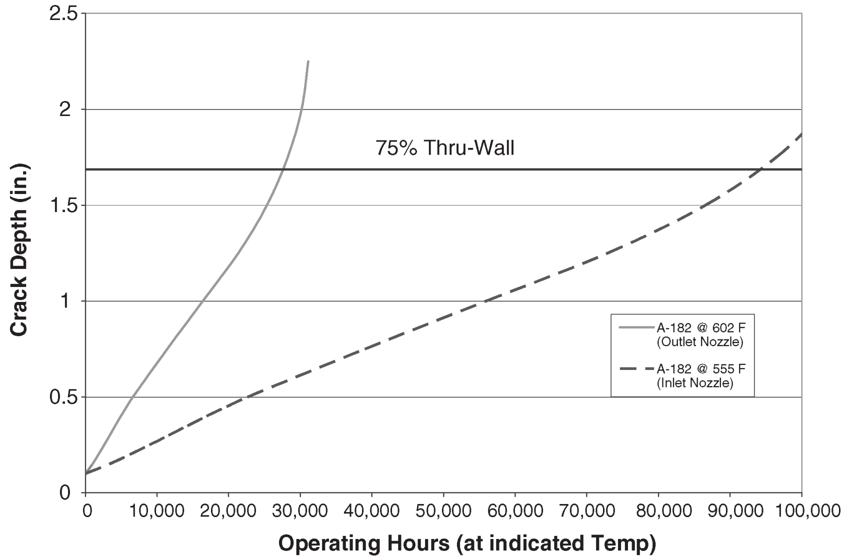

FIG. 44.16 CRACK GROWTH RATE PREDICTIONS FOR CIRCUMFER-ENTIAL CRACKS IN RPV TOP-HEAD NOZZLE AT VARIOUS ASSUMED OPERATING TEMPERATURES INITIAL CRACK ASSUMPTION = 30° THROUGH-WALL CRACK AT MAXIMUM STRESS AZIMUTH IN HIGH ANGLE NOZZLE.

a 6-to-1 aspect ratio crack in a large-diameter RPV inlet/outlet nozzle, ranging in depths from 0.1 in. to 2.2 in. The nozzle was conservatively assumed to have a large, inside surface repair, and the crack was assumed to reside in the center of that repair (i.e., in the most unfavorable residual stress region of the weld). The predicted crack growth in this case is fairly rapid for a typical outlet nozzle temperature, 602°F, propagating to 75% through-wall (the upper bound of ASME Section XI allowable flaw sizes in piping) in about 3 years. Conversely, if no weld repair were assumed, little or no crack growth would be predicted over the plant lifetime. For this same crack, including the effect of the repair, the predicted time for a 0.1 in. deep crack to grow to 75% through-wall at a typical inlet nozzle temperature (555°F) is about 11 years.

The strong effect of operating temperature is apparent in both crack growth analyses. The outlet nozzle analysis also demonstrates the detrimental effect of weld repairs that were performed during construction at some plants.

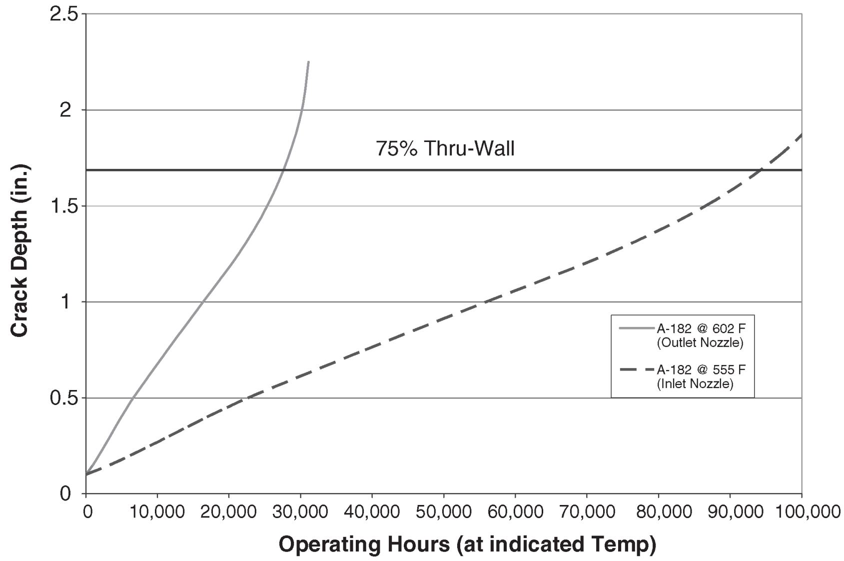

FIG. 44.17 CRACK GROWTH RATE PREDICTIONS FOR CIRCUMFERENTIAL CRACKS IN RPV MAIN COOLANT LOOP DISSIMILAR METAL NOZZLE BUTT WELD AT OPERATING TEMPERATURES TYPICAL OF REACTOR INLET AND OUTLET NOZZLES INITIAL CRACK ASSUMPTION = 0.1″ × 0.6″ INSIDE SURFACE CRACK AT MAXIMUM STRESS AZIMUTH IN NOZZLE WITH ASSUMED INSIDE SURFACE FIELD REPAIR.

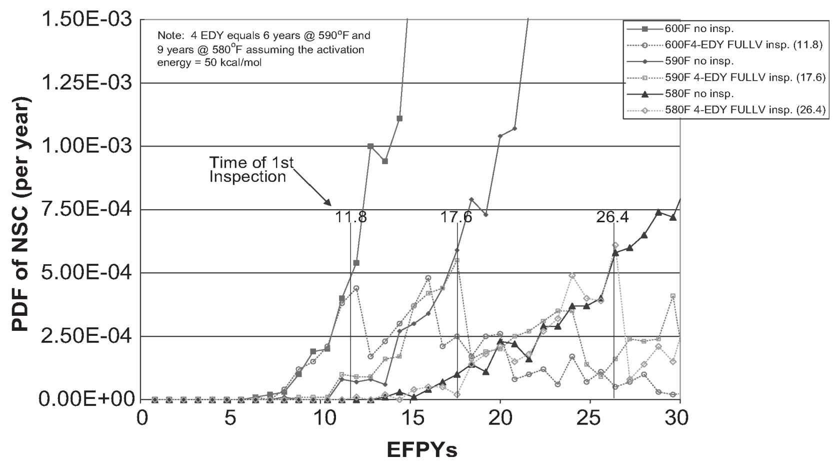

FIG. 44.18 PROBABILITY OF NOZZLE FAILURE (NSC) AS A FUNCTION OF VARIATIONS IN TOP-HEAD TEMPERATURE AND INSPECTION INTERVALS

44.7.3 Probabilistic Analysis

Because of the large degree of statistical scatter in both the crack initiation and crack growth behavior of PWSCC in alloy 600 base metal and associated weld metals, probabilistic fracture mechanics (PFM) analyses have been used to characterize the phenomenon in terms of the probabilities of leakage and failure [49] for RPV top head nozzles. The analysis incorporates the following major elements:

(a) computation of applied stress intensity factors for circumferential cracks in various nozzle geometries as a function of crack length and stresses
(b) determination of critical circumferential flaw sizes for nozzle failure
(c) an empirical (Weibull) analysis of the probability of nozzle cracking or leakage as a function of operating time and temperature of the RPV head
(d) statistical analysis of PWSCC crack growth rates in the PWR primary water environment as a function of applied stress intensity factor and service temperature
(e) modeling of the effects of inspections, including inspection type, frequency, and effectiveness

A series of PFM analysis results is presented in [49], which covers a wide variety of conditions and assumptions. These include base cases, with and without inspections, and sensitivity studies to evaluate the effects of various statistical and deterministic assumptions. The model was benchmarked with respect to field experience, considering the occurrence of cracking and leakage and of circumferential cracks of various sizes. The benchmarked parameters were then used to evaluate the effects of various assumed inspection programs on probability of nozzle failure and leakage in actual plants. A sample of the results is presented in Figs. 44.18 and 44.19.

Figure 44.18 shows the effect of inspections on probability of nozzle failure (Net Section Collapse, or ejection of a nozzle) for head operating temperatures ranging from 580°F to 600°F. A no-inspection curve is shown for each temperature. Runs were then made assuming NDE inspections of the nozzles. Inspections were assumed to be performed at intervals related to head operating temperature (more frequent inspections for higher head temperatures,

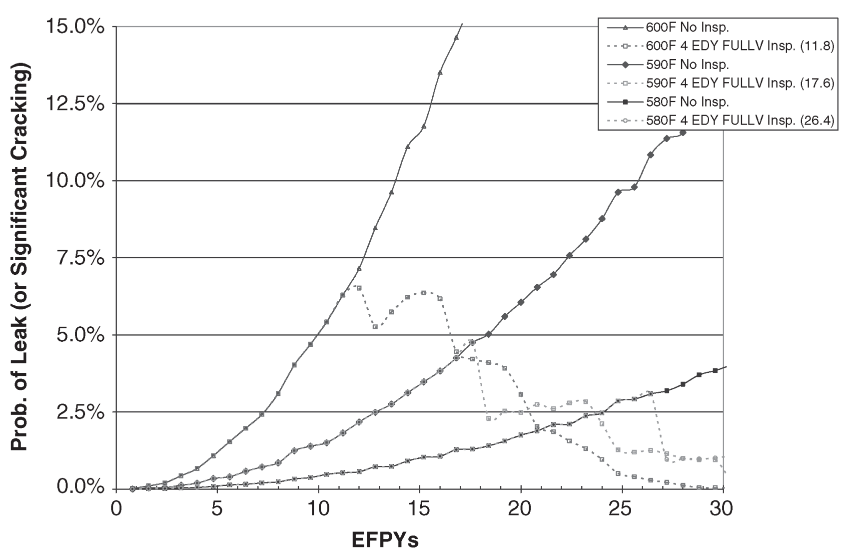

FIG. 44.19 PROBABILITY OF NOZZLE LEAKAGE AS A FUNCTION OF VARIATIONS IN TOP-HEAD TEMPERATURE AND INSPECTION INTERVALS

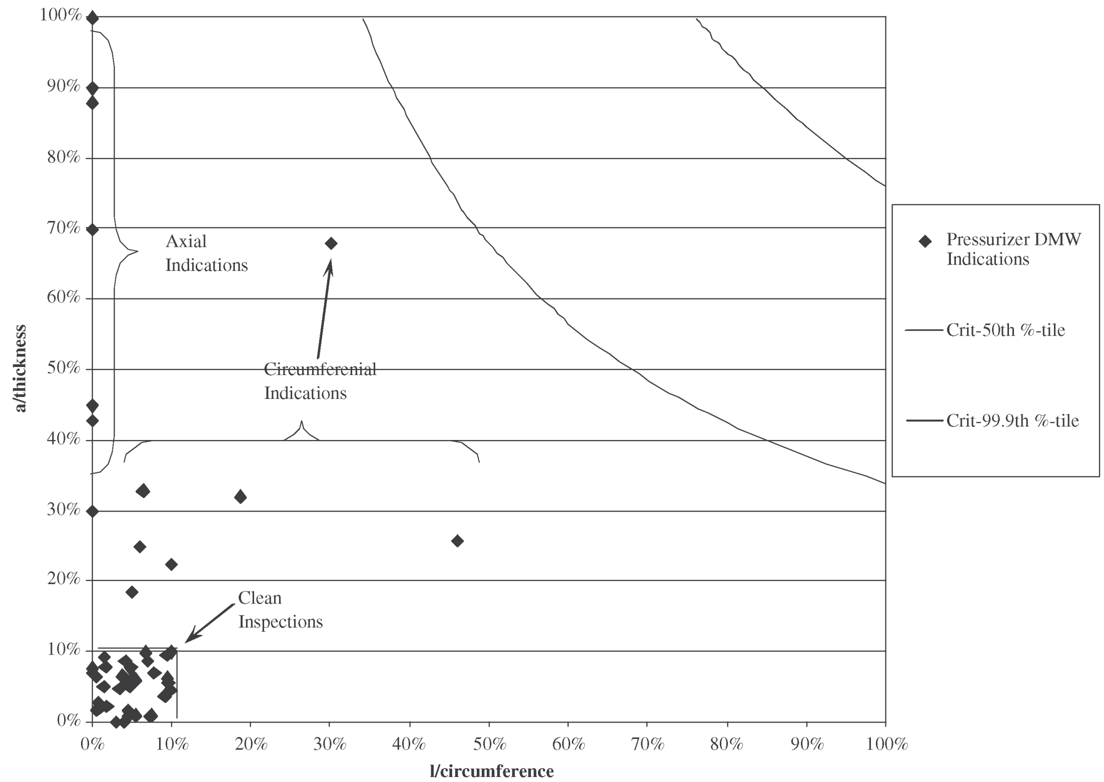

FIGURE 44.19A PRESSURIZER DISSIMILAR METAL BUTT WELD FLAW INDICATIONS COMPARED TO CRITICAL FLAW SIZE PROBABILITY ESTIMATES

less frequent for lower temperatures). It is seen from the figure that the assumed inspection regimen is sufficient to maintain the nozzle failure probability (per plant year) below a generally accepted target value of 1×10^{-3} for loss of coolant accidents due to nozzle ejection.

Figure 44.19 shows similar results for the probability of leakage from a top-head nozzle. It is seen from this figure that the same assumed inspection regimen maintains the probability of leakage at or about 6% for the cases analyzed. Analyses similar to those reported in Figs. 44.18 and 44.19 have been used, in conjunction with deterministic analyses, to define an industry-recommended inspection and corrective action program for PWR top heads that results in acceptable probabilities of leakage and failure. This work also constituted the basis for the inspection requirements incorporated in ASME Code Case N-729-1 [63].

Similar probabilistic analyses have been performed for PWSCC susceptible butt welds in pressurizer nozzles, as part of the effort documented in MRP-216 [59]. Analyses established the current expected flaw distribution based on pressurizer nozzle DMW inspections to date, (Table 44.1), estimates were made of the probability of cracking versus flaw size, and of crack growth rate versus time. A plot of the flaw indications found to date, in terms of crack length as percentage of circumference (abscissa) versus crack depth as percentage of wall thickness (ordinate) is illustrated in Figure 44.19a. Axial indications plot along the vertical axis (l/circumference = 0) in this plot, with leaking flaws plotted at a/t = 100%. Circumferential indications plot at non-zero values of l/circumference, at the appropriate a/t. Clean inspections are plotted randomly in a 10% box near the origin, to give some indication of inspection uncertainty. Also shown on this plot are loci of critical flaw sizes based on an evaluation of critical flaw sizes presented in Ref. [59]. 50th and 99.9th percentile plots are shown. It is seen from this figure that all of the flaw indications detected were far short of the sizes needed to cause a rupture. The probabilistic analysis also addressed the small but finite probability that larger flaws may exist in uninspected nozzles, plus the potential for crack growth during future operating time until all the nozzles are inspected (or mitigated) under MRP-139 [58] guidelines.

44.8 REPAIRS

When cracking or leakage is detected in operating nuclear power plant pressure boundary components, including the reactor vessel, repair or replacement may be performed in accordance with ASME BPVC Section XI [51]. Section XI specifies that the flaws must be removed or reduced to an acceptable size in accordance with Code-accepted procedures. For PWSCC in RPV alloy 600 components, several approaches have been used.

44.8.1 Flaw Removal

For relatively shallow or minor cracking, flaws may be removed by minor machining or grinding. This approach is permitted by the ASME Code to eliminate flaws and return the component to ASME Code compliance. However, this approach generally does not eliminate the underlying cause of the cracking. There will still be susceptible material exposed to the PWR environment that caused the cracking originally, and in some cases the susceptibility might be aggravated by surface residual stresses caused by the machining or grinding process. Simple flaw removal is thus not considered to be a long-term repair, unless combined with some other form of mitigation. However, in the short term, for example, where future component replacement is planned, it may be a viable approach for interim operation.

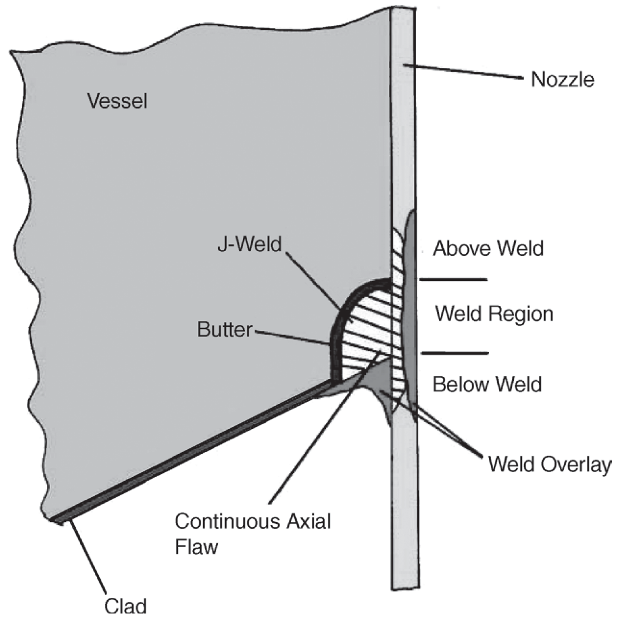

FIG. 44.20 SCHEMATIC OF RPV TOP-HEAD NOZZLE FLAW EMBEDMENT REPAIR

44.8.2 Flaw Embedment

Surface flaws are much more significant than embedded flaws from a PWSCC perspective, because they continue to be exposed to the PWR primary water environment that caused the crack and that can lead to continued PWSCC flaw growth after initiation. Accordingly, one form of repair is to embed the flaw under a PWSCC-resistant material. Figure 44.20 shows an embedment approach used by one vendor to repair PWSCC cracks or leaks in top-head nozzles and welds. The PWSCC-susceptible material, shown as the cross-hatched region in the figure, is assumed to be entirely cracked (or just about to crack). PWSCC-resistant material, typically alloy 52 weld metal, is deposited over the susceptible material. The assumed crack is shown to satisfy all ASME BPVC Section XI flaw evaluation requirements, in the absence of any growth due to PWSCC, since the crack is completely isolated from the PWR environment by the resistant material. Note that the resistant material in this repair must overlap the susceptible material by enough length in all directions to preclude new cracks growing around the repair and causing the original crack to be reexposed to the PWR environment. Although this repair approach has been used successfully in several plants, there have been many incidents in which nozzles repaired by this approach during one refueling outage have been found to be leaking at the subsequent outage. These occurrences were attributed to lack of sufficient overlap of the repair, because it is sometimes difficult to distinguish the exact point at which the susceptible material ends (for instance the end of the J-groove weld butter and the beginning of the RPV cladding in Fig. 44.20).

44.8.3 Weld Overlay

Another form of repair that has been used extensively to repair cracked and leaking pipe welds is the weld overlay (WOL). Illustrated schematically in Fig. 44.21, WOLs were first conceived in the early 1970s as a repair for IGSCC cracking and leakage in BWR main coolant piping. Over 500 such repairs have been applied in BWR piping ranging from 4 in. to 28 in. in diameter, and some weld overlay repairs have been in service for over 20 years, with no evidence of any resumption of the IGSCC

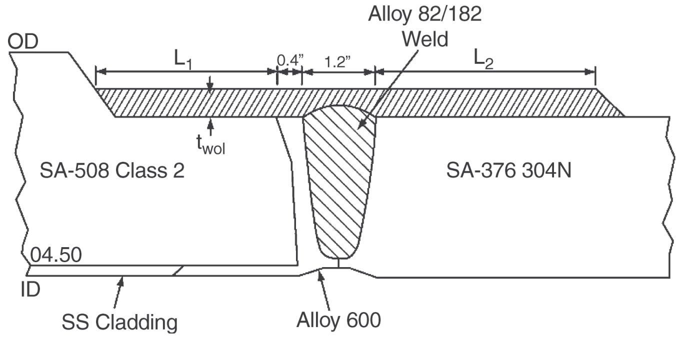

FIG. 44.21 SCHEMATIC OF WELD OVERLAY REPAIR APPLIED TO RPV OUTLET NOZZLE

problem. Although WOLs, shown in Fig. 44.21, do not eliminate the PWSCC environment from the flaw as in the flaw embedment process, the repair has been shown to offer multiple improvements to the original pipe welds, including the following:

(a) structural reinforcement
(b) resistant material
(c) favorable residual stress reversal

Weld overlays also offer a significant improvement in inspection capability, because once a weld overlay is applied, the required inspection coverage reduces to just the weld overlay material plus the outer 25% of the original pipe wall, often a much easier inspection than the original dissimilar metal weld (DMW) inspection.

Weld overlay repairs have been recognized as a Code-acceptable repair in an ASME Section XI Code Case [52] and accepted by the U.S. NRC as a long-term repair. They have also been used, albeit less extensively, to repair dissimilar metal welds at nozzles in BWRs.

The weld overlay repair process was first applied to a PWR large-diameter pipe weld (on the Three Mile Island 1 pressurizer to hot-leg nozzle) in the fall of 2003. Since that time, as part of the MRP-139 inspection effort described in para. 44.5.6, over 200 weld overlays have been applied to pressurizer nozzle dissimilar metal butt welds. Part of the reason for this trend is that many pressurizer nozzles were unable to be volumetrically inspected to achieve the required examination coverage in their original configuration. By applying weld overlays, in addition to mitigating the welds, their inspectability was enhanced such that post overlay ultrasonic exams could be performed in accordance with applicable requirements. Technical justification for the WOL process as a long-term repair is documented in Ref. [53]. Requirements for weld overlays in PWR systems, including their use as mitigation as well as repair, is documented in Ref. [60].

44.8.4 Weld Replacement

Finally, the flawed weld may be replaced in its entirety. In PWR top-head nozzles, this process has been implemented extensively by relocating the pressure boundary from the original PWSCC-susceptible J-groove weld at the inside surface to a new weld at the midwall of the RPV head (see Fig. 44.22). With this repair approach, the PWSCC-susceptible portion of the original J-groove weld and buttering is left in the vessel, but it is no longer part of the pressure-retaining load path for the nozzle. The lower portion of the original nozzle is first removed by machining to a horizontal elevation above the J-groove weld (left-hand side of Fig. 44.22). A weld prep is produced on the bottom of the remaining portion of

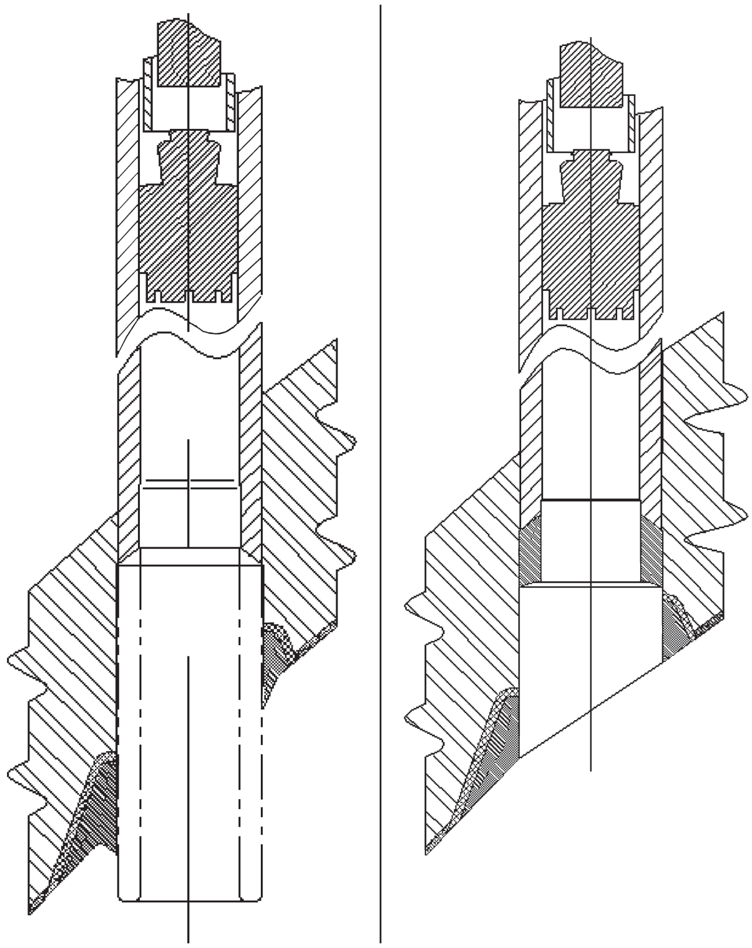

FIG. 44.22 SCHEMATIC OF RPV TOP-HEAD NOZZLE WELD REPLACEMENT REPAIR

the nozzle, and a new, horizontal weld is made between the original nozzle and the bore of the RPV head (righthand side of Fig. 44.22). The new weld is made with PWSCC-resistant material (generally alloy 52 weld metal), and the surface of the weld is machined for NDE. The repair process still leaves some portion of the original PWSCC-susceptible alloy 600 nozzle in place, potentially in a high residual stress region at the interface with the new weld. However, a surface treatment process, such as roll peening or burnishing, has been applied to this interface in many applications to reduce potential PWSCC concerns. Experience with this repair process has been good, in terms of subsequent leakage from repaired nozzles, and in most cases the repair need only survive for one or two fuel cycles, because, once PWSCC leakage is detected in an RPV head, common industry practice has been to schedule a future head replacement (not because of the repaired nozzle, but because of concerns that other nozzles are likely to be affected by the problem leading to costly future inspections, repairs, and outage extensions).

44.9 REMEDIAL MEASURES

44.9.1 Water Chemistry Changes

Three types of water chemistry changes that could affect the rate of PWSCC are zinc additions to the reactor coolant, adjustments to hydrogen concentration, and adjustments to lithium concentration and pH. The factors are described below.

(a) *Zinc Additions to Reactor Coolant.* Laboratory tests indicate that the addition of zinc to reactor coolant significantly slows down the rate of PWSCC initiation, with the improvement factor increasing as the zinc concentration increases [29]. The improvement factor (slowdown in rate of new crack initiation) shown by tests varies from a factor of two for 20 ppb zinc in the coolant to over a factor of ten for 120 ppb zinc. The effect of zinc on crack growth rate is not as certain, with some tests indicating a significant reduction in crack growth rate but others indicating no change. Further testing is underway under EPRI sponsorship (as of 2004) to clarify the effects of zinc on crack growth rate. As of mid-2004, evaluation of plant data, especially the data for a two-unit station with PWSCC at dented steam generator tube support plates, is encouraging but not conclusive with regard to whether use of zinc is reducing the rate of PWSCC. The uncertainty is the result of changes in inspection methods simultaneously with changes in zinc concentration.

(b) *Adjustments of Hydrogen Concentration.* The EPRI PWR Primary Water Chemistry Guidelines require the hydrogen concentration in the primary coolant to be kept between 25 and 50 cc/kg [28]. As discussed in the Guidelines and summarized above in para. 44.3.4, the rate of PWSCC initiation and rate of PWSCC crack growth both seem to be affected by the hydrogen concentration, with lower concentrations being more aggressive at lower temperature and higher concentrations at higher temperature. Depending on the plant situation as far as which parts are at most risk of PWSCC, and depending on the temperature at those parts, there may be some benefit, such as an improvement factor of about 1.2, in operating at hydrogen concentrations at either end of the allowed range. In the longer term, increased benefit may be achieved by operating slightly outside of the allowed range (e.g., at 60 cc/kg), although this will require confirmation that the change does not result in some other undesirable effects.

(c) *Adjustments of Lithium Concentration and pH.* As discussed in para. 44.3.4, some tests indicate that the rate of PWSCC initiation is increased by increases in lithium concentration and pH, e.g., by factors ranging from about 1.15 to 2.0. On the other hand, increases in lithium and pH provide proven benefits for reducing the potential harmful deposit buildup on fuel cladding surfaces and for reducing shutdown dose rates [28]. Based on these opposing trends, plants can select a lithium/pH regime that best suits their needs, i.e., does not involve substantial risks of aggravating PWSCC, while still providing benefits for reducing fuel deposits and shutdown dose rates. When evaluating the possible risks to PWSCC of increasing lithium and pH, it should be noted that crack growth rate tests show no harmful effect while crack initiation tests do. The data from crack growth rate tests are considered to be more reliable, and it is recommended that they be given greater weight than the results from crack initiation tests. An additional consideration is that the use of zinc can provide a stronger benefit than the possible deficit associated with increases in lithium and pH, and, thus, can make use of a combined zinc adjustment and increase in lithium and pH attractive.

44.9.2 Temperature Reduction

To date, a main remedial measure applied in the field for RPV top-head PWSCC has been modification of the reactor internals package to increase bypass flow through the internals flange region and, thereby, reduce the head temperature. The lower head temperature is predicted to reduce the rates of crack initiation and growth based on the thermal activation energy model, as shown in Table 44.1. However, experience in France suggests that PWSCC may occur at head temperatures close to the reactor cold-leg temperature. This is especially significant given PWSCC of two South Texas Project Unit 1 BMI nozzles at a temperature of about 565°F. The South Texas Project experience shows that materials and fabrication-related factors can result in PWSCC at temperatures lower than otherwise expected.

44.9.3 Surface Treatment

EPRI has sponsored tests of a range of mechanical remedial measures for PWSCC of alloy 600 nozzles. Results of these tests were reported by Rao at the Fontevraud 5 Symposium [54]. The remedial measures test program consisted of soliciting remedial measures from vendors, fabricating full-diameter and wall-thickness ring specimens from archive CRDM nozzle material, installing specimens in rings that locked in high residual stresses on the specimen inside surface, applying the remedial measures to the stressed surface, and then testing the specimens in doped steam with hydrogen overpressure at 400°C (750°F). The specimens were removed from the autoclave at intervals and inspected for SCC. A complicating factor in interpreting the test results is that not all of the specimens were fabricated from the same heat of material. Therefore, there were differences in material PWSCC susceptibility in addition to differences in remedial measure effectiveness. The methods used to correct for differences in specimen PWSCC susceptibility are discussed in the paper.

The remedial measures fell into three main effectiveness groups.

(a) most effective

(1) waterjet conditioning
(2) electro mechanical nickel brush plating
(3) shot peening

(b) intermediate effectiveness

(1) electroless nickel plating
(2) GTAW weld repair
(3) laser weld repair

(c) least effective

(1) EDM skim cutting
(2) laser cladding
(3) flapper wheel surface polishing

As of May 2005, it is not believed that any of these remedial measures had actually been applied to a reactor vessel in the field.

44.9.4 Stress Improvement

To mitigate against the IGSCC problem in BWR piping, many plants implemented residual stress improvement processes. These were performed both thermally (induction heating stress improvement or IHSI) and by mechanical means (mechanical stress improvement process or MSIP). As described above, residual stresses play a major role in susceptibility to both IGSCC and PWSCC, because large piping butt welds tend to leave significant residual stresses at the inside surfaces of the pipes, especially when field repairs were performed during construction. Both stress improvement processes have been demonstrated to reverse the unfavorable residual stresses, leaving compressive stresses on the inside surface of the pipe, which is exposed to the reactor environment. MSIP has also been applied to PWSCC-susceptible butt welds in PWR piping, primarily dissimilar metal welds at vessel nozzles, such as the V.C. Summer outlet nozzle cracking problem described above. As long as the stress improvement process is applied relatively early in life, when cracking has not initiated or grown to significant depths, it clearly constitutes a useful remedial measure that can be applied to vessel nozzles, eliminating one of the major factors that contribute to PWSCC.

One of the benefits of the weld overlay process described above to repair PWSCC-cracked butt welds is that it reverses the residual stress pattern in the weld, resulting in compressive stresses on the inside surface. Thus, a novel mitigation approach that is being explored at several plants is the application of weld overlays preemptively, before cracking is discovered. Applying a preemptive WOL in this manner produces the same remedial benefits described above for the stress improvement processes, but also places a PWSCC-resistant structural reinforcement on the outer surface of the pipe. So, if the favorable residual stresses were to relax in service, or for some reason be ineffective in arresting the PWSCC phenomenon, the PWSCC-resistant overlay would still provide an effective barrier against leakage and potential pipe rupture. Moreover, the revised inspection coverage requirements specified for WOLs apply to such preemptive overlays, providing the added benefit of enhanced inspectability [52].

44.9.5 Head Replacement

The most obvious way to address RPV top-head cracking issues is head replacement. Approximately one-third of operating PWRs in the United States have replaced their heads or have scheduled head replacements in the near future. Such head replacements take advantage of the lessons learned to date regarding the PWSCC phenomenon, and the new heads are generally produced so as to eliminate all PWSCC-susceptible materials, replacing them with resistant materials (alloy 690 and associated weld metals alloys 52 and 152). RPV head replacement is a key aspect of strategic planning to address the alloy 600 problem in PWRs, and is performed as part of a coordinated alloy 600 maintenance program that addresses steam generator, pressurizer, and piping issues as well as the RPV.

44.10 STRATEGIC PLANNING

Within constraints posed by regulatory requirements, utilities are free to develop a strategic plan that ensures a low risk of leakage, ensures an extremely low risk of core damage, and results in the lowest net present value (NPV) cost consistent with the first two criteria. Development of a strategic plan for RPV top-head nozzles was described by White, Hunt, and Nordmann at the 2004 ICONE-12 conference [55]. The strategic planning process was based on life cycle management approaches and NPV economic modeling software developed by EPRI [56,57].

The main steps in the strategic planning process are as follows:

(a) predicting time to PWSCC
(b) assessing risk of leaks
(c) assessing risk of rupture and core damage due to nozzle ejection
(d) assessing risk of rupture and core damage due to boric acid wastage
(e) identifying alternative life cycle management approaches
(f) evaluating economically the alternative management approaches

While the paper and following discussion are based on RPV top-head nozzles, the same basic approach can be applied to BMI nozzles and butt welds.

44.10.1 Predicting Time to PWSCC

Predictions of the time to PWSCC crack initiation are described in para. 44.7.1. The predictions are typically based on a statistical distribution such as a two-parameter Weibull or lognormal model. Predictions are most accurate if based on plant-specific repeat inspections showing increasing numbers of cracked nozzles. If such data are not available, then predictions are typically based on data for other similar plants (e.g., design, material, operating conditions) with corrections for differences in operating time and temperature.

44.10.2 Assessing Risk of Leaks

The risk of leakage at a particular point in time (typically refueling outage number) is typically determined by a probabilistic (Monte-Carlo) analysis using the distribution of predicted time to crack initiation (para. 44.7.1), crack growth (para. 44.7.2), and other probabilistic modeling techniques (para. 44.7.3). The probabilistic analysis should include a sensitivity study to identify the most important analysis input parameters, and these important parameters should be reviewed to ensure that they can be substantiated by available data.

44.10.3 Assessing Risk of Rupture and Core Damage Due to Nozzle Ejection

The risk of nozzle ejection (net section collapse) is determined using methods such as described in para. 44.6.2.

44.10.4 Assessing Risk of Rupture and Core Damage Due to Boric Acid Wastage

The risk of failure of the carbon or low-alloy steel reactor vessel head by boric acid wastage is determined using methods such as described in para. 44.6.3.

44.10.5 Identifying Alternative Life Cycle Management Approaches

An important step in developing a life cycle management plan is to identify the alternative approaches that can be considered. These alternatives can include the following:

(a) continue to inspect and repair indefinitely without applying remedial measures.
(b) apply remedial measures, such as lowering the vessel head temperature by increasing bypass flow through the vessel internals flange, adding zinc to the primary coolant, and water-jet conditioning the wetted surface of nozzles and welds to remove small flaws and leave the material surface with a compressive residual stress.
(c) replace the vessel head as quickly as possible.
(d) replace the vessel head shortly after detecting the first PWSCC cracks.
(e) use other approaches identified.

Each of these alternatives must be studied to determine the difficulty of application, the likely effectiveness, and the effect of the change on required inspections. For example, head replacement may involve the need to cut an access opening in the containment structure or to procure a new set of CRDMs to allow the head changeout to be performed quickly, so as to not adversely affect the refueling outage time. If openings must be cut in containment, consideration should also be given to the possible need to cut other openings in the future, such as for steam generator or pressurizer replacements. Consideration must also be given to the disposal of a head after it is replaced.

44.10.6 Economic Evaluations of Alternative Management Approaches

Most life cycle management evaluations include economic analyses to determine the NPV cost of each alternative. The NPV cost is the amount of money that is required today to pay all predicted future costs, including the effects of inflation and the discount rate. Inputs to an LCM economic analysis typically include the following:

(a) costs of planned preventive activities including inspections, remedial measures, and replacements.
(b) predicted failure mechanisms (e.g., cracks, leaks, and rupture) and failure rates.
(c) costs for corrective maintenance in the event of a failure including the cost to make the repair, the estimated value of lost production, and an allowance for consequential costs such as increased regulatory scrutiny. Consideration should be given to the fact that a major incident such as the Davis-Besse RPV head wastage can result in lost production and consequential costs far higher than the cost to replace the affected component.

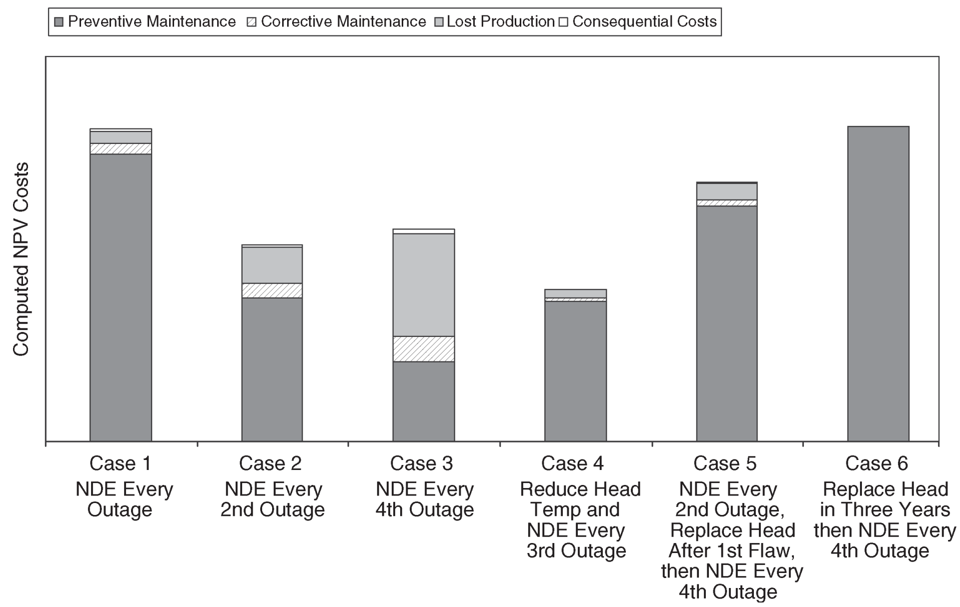

FIG. 44.23 TYPICAL RESULTS OF STRATEGIC PLANNING ECONOMIC ANALYSIS FOR RPV HEAD NOZZLES

Figure 44.23 shows typical results of a strategic planning analysis with economic modeling.

The final step in the economic evaluation is to review the predictions in light of other plant constraints, such as planned plant life, potential power uprates, budget constraints, and the availability of replacement heads. In many cases, the alternative with the lowest predicted NPV cost may not represent the best choice.

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